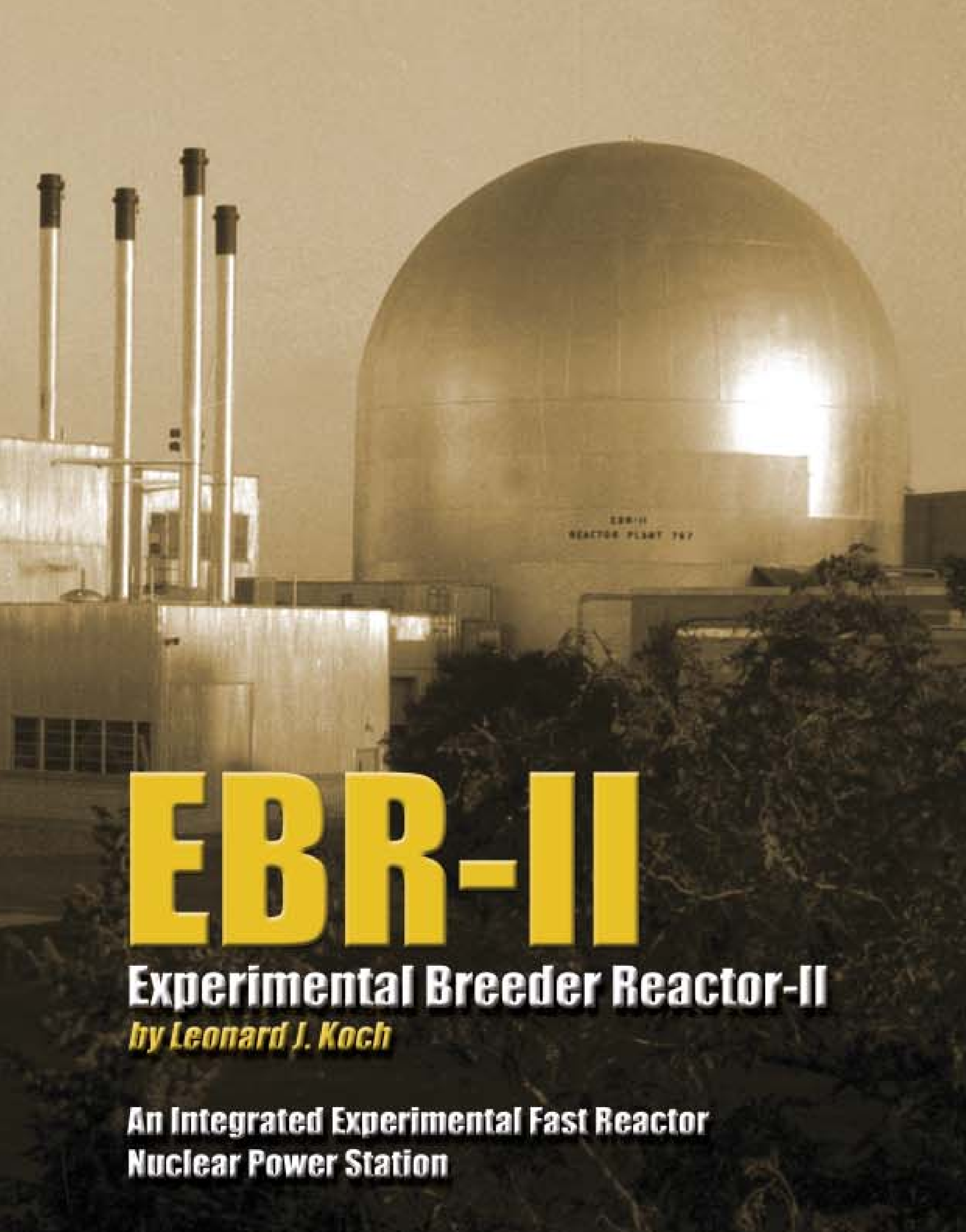

EBR-II

Experimental Breeder Reactor-II

by Leonard J. Koch

An Integrated Experimental Fast Reactor Nuclear Power Station

Experimental Breeder Reactor-II (EBR-II)

An Integrated Experimental Fast Reactor Nuclear Power Station

By
Leonard J. Koch

Authorized By
Argonne National Laboratory

This book is dedicated to my wife Rosemarie, whose support and patience permitted me to spend most of my more than 60 years of professional life in the pursuit and application of advanced cutting edge technology. First in the development of internal combustion engines and then in the development of nuclear power including its introduction, application, and hopefully the realization of its real potential. Most recently it has included the writing of this book, which is intended to assist in realizing that potential.

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FORWARD

The Experimental Breeder Reactor-II (EBR-II) and the associated pyrometallurical processing facility were remarkable engineering achievements of the last century. We are fortunate that the manager of the project, Leonard Koch, has written this first-hand account of the development, design, construction, and initial operation of this facility, which has contributed to the foundation of the knowledge for all fast reactors in the world. It captures the total process involved, including the very beginning when the basic concept resided in the minds of brilliant men. They had discovered the enormous amount of energy that could be produced by nuclear fission and had conceived of a concept for the controlled extraction of that energy. They had correctly recognized that it would be necessary to recycle the nuclear fuel many times to use it effectively and had calculated that this could only be achieved with fast neutrons.

Leonard Koch's early career at Argonne National Laboratory was exemplary as well. He was on the original team that designed and constructed Experimental Breeder Reactor-I (EBR-I). He participated when EBR-I generated the world's first useful electricity from nuclear power on December 20, 1951. His name appears on the EBR-I wall with Walter Zinn and his other colleagues.

This book on EBR-II is a detailed presentation of the design and construction of EBR-II, which is augmented with numerous original drawings and photographs, and thus will be of great use to the designers of future fast reactors. Leonard Koch was careful to explain why certain design choices were made while others were rejected. He also offers a section on how he believes future sodium cooled fast reactors should be designed, based on the experience gained with EBR-II.

Of general and historical interest, this book includes an appendix that traces the lineage of EBR-II, including original memos and meetings notes, beginning with Enrico Fermi and Walter Zinn and progressing to the formation of the EBR-II project.

We owe a debt of gratitude to Leonard Koch for interrupting his retirement to give us this excellent account of one of Argonne's greatest achievements. This book will be of great value for generations to come.

HERMANN A. GRUNDER, DIRECTOR
ARGONNE NATIONAL LABORATORY

PREFACE

The discovery of nuclear fission and of the self-sustaining, controlled fission in a nuclear reactor led to the development of the atomic bomb and to the recognition of the tremendous potential of this new highly-concentrated energy source. The magnitude of the energy potentially available from application of Einstein's theory was awesome; (i.e., while 1 pound of coal can produce approximately 3 kilowatt-hours of thermal energy, 1 pound of uranium can produce more than 10 million kilowatt-hours). Virtually all of the power plants that now operate in the world (all of them in the United States) extract less than 1 percent of this potential energy. Therefore, we now produce about 100,000 kilowatt-hours of heat energy from each pound of uranium we use (use, not consume). This, of course, is much more energy than we extract from a pound of coal, but we "waste" about 9,900,000 kilowatt-hours of the energy potential in each pound of uranium. A primary objective of the EBR-II project, which is described in this book, was to develop and demonstrate the technology that can provide the capability to extract much more of that energy.

The discovery of nuclear fission was greeted by intense enthusiasm by the technical community and the general public; perhaps over enthusiasm, which is not uncommon for new discoveries. "Atomic power" would become the universal energy source powering automobiles, airplanes, space travel, and producing "electricity too cheap to meter."

In the early 1950s, it was established that nuclear power should be a potential energy source for electric power generation and for submarine propulsion. The former, because it was demonstrated that large quantities of energy could be produced in production reactors, and it would be necessary only to increase the working temperature to accomplish energy conversion to electricity. For submarines, the unique advantage of a non-air burning requirement would permit virtually unlimited submerged operation of the ship at all power levels.

With general agreement that these applications should be developed, attention was focused on the options available and the capability to produce the desired results. The U.S Atomic Energy Commission initiated development of a variety of potential reactor concepts for electric power generation (and two basic concepts for submarine propulsion). EBR-II was one of the power reactor concepts selected for development. Two light water power reactor concepts were also selected—pressurized water and boiling water.

Light water reactors were given initial emphasis because they were considered "technologically easier" and could be developed more quickly, but EBR-II was based on the technology which would (and should) make nuclear power a very long-term energy source with a virtually unlimited supply of fuel.

Depending on the type of nuclear reactor and the fuel system used, the number of neutrons produced and their utilization will determine the conversion ratio of uranium-238 to plutonium. For example, in a uranium-fueled light water thermal neutron power reactor (virtually the only type of power reactor used in the United States), about one atom of plutonium is produced for each two atoms of uranium-235 fissioned while in a plutonium fueled fast neutron power reactor, about three atoms of plutonium can be produced for each two atoms of plutonium fissioned. The former type of reactor is known as a converter while the latter is termed a breeder (because it produces more plutonium than it consumes).

These basic characteristics of nuclear power reactors result in a utilization of less than 1 percent of the energy content of natural uranium in thermal converter reactors, when taking into account the natural uranium used to produce the “low enrichment” fuel for light water reactors; while the potential utilization of virtually 100 percent of the energy content of natural uranium is achievable in fast breeder reactors. In addition, this same level of uranium utilization can be achieved in fast reactors with “depleted uranium,” (i.e., natural uranium from which much of the uranium-235 has been extracted to produce enriched uranium with uranium-235 content from about 1 percent to 90 percent for uses, including military and fuel for light water reactors). Achieving a high level of uranium utilization was basic to the Argonne concept of nuclear power. Although the Laboratory was involved in the development of other nuclear power concepts, the primary research and development efforts were directed to support this basic philosophy. The Argonne concept is based on the premise that nuclear power should be produced by “burning” natural uranium (depleted uranium as long as it is available) and is accomplished by converting uranium-238 to plutonium in fast reactors. In this process, plutonium functions as the catalyst for consuming uranium-238 and the reactor is fueled with uranium-238.

Processing nuclear fuel to properly produce the needed recycle products is a basic requirement for operation of a fast neutron power reactor system. One objective of the EBR-II was to demonstrate the integrated operation of the reactor power system and the fuel cycle system as a closed energy supply system.

Demonstrating the feasibility of a closed energy supply system was believed necessary to resolve many uncertainties: Could fuel recycle be accomplished in such a manner that the total fuel inventory in the system would be of acceptable size? Could the buildup of heavy isotopes (uranium-236, uranium-237, plutonium-240, plutonium-241, other actinides, etc.) be accommodated in a closed fuel cycle system? Since pyrometallurgical processes appeared to have the potential for application in such a system, but had the disadvantage of incomplete decontamination of the fuel, could a realistic closed fuel system accommodate the recycle of highly-radioactive fuel? The EBR-II program attempted to address these questions. In the process, the program goal included advancing the technology of fast neutron power reactors for long-term generation of electricity.

From the beginning, it was recognized that fast reactors would be significantly different from thermal (neutron moderated) reactors. The much smaller neutron cross-sections, both fission and capture, led to an entirely different geometry and design of these reactors. The fast reactor required high fuel density and relatively high fuel enrichment to achieve criticality in a fast neutron environment in which the fission cross-section is small. On the other hand, a broad choice of materials was permissible for reactor structures and coolant, also because of the small neutron capture cross-sections. Fast reactors were different and would continue to be different as they evolved.

Fast reactors are relatively insensitive to fission product buildup and their effect on reactivity of the reactor. This unique characteristic of fast reactors to tolerate fission product buildup made it feasible to incorporate into the EBR-II a fuel recycle process that did not remove all of the fission products. Although these fission products capture neutrons, the impact on the total neutron balance is insignificant.

A unique aspect of a liquid metal coolant is its high boiling temperature at atmospheric pressure. As a result, the cooling systems operate at essentially atmospheric pressure and the loss of coolant accident assumes an entirely different dimension. This characteristic permitted the EBR-II concept to include submerging the reactor and primary coolant system in the primary coolant and to devise the many benefits that resulted. Such a concept would be impossible in a high pressure cooling system such as water.

Although it was not originally recognized as a significant characteristic, the EBR-II concept is responsive to many of the concerns about nuclear fuel cycles which have developed since EBR-II. Weapons proliferation has become a major concern related to commercial nuclear power. The EBR-II fuel is naturally proliferation-resistant at all times in the cycle. New reprocessed fuel is still highly radioactive and the fissionable isotope, whether plutonium or uranium-235, is never cleanly separated from this highly-radioactive material. This results in an unattractive source of weapons material. As operations continue and the fuel continues to be recycled it becomes even less attractive as a weapons material source because of the natural buildup of the higher isotopes of plutonium and other actinides.

The use of on-site fuel recycle as pioneered by EBR-II, eliminates the shipment of irradiated material, thus making it far less accessible to would-be proliferators, and decreases the opportunity for theft or misappropriation. The absence of the need for shipment through public space eliminates the public safety concerns about shipping highly-radioactive or toxic materials.

All of these considerations contributed to the genesis of this book, but by far the most compelling was the desire to improve the utilization of uranium.

I have been driven by the conviction that much more than 1 percent of the energy contained in uranium must be utilized if nuclear power is to achieve its real long-term potential. It was correctly established by Enrico Fermi and others more than 50 years ago that this could be accomplished in fast reactors if the fuel was properly reprocessed and recycled repeatedly to extract that available energy (rather than store it as waste).

EBR-II was an early attempt to establish the technology needed to exploit this science. It was partially successful. I was among those that expected the pursuit of this needed technology would continue. Although this has not occurred, I am still of the conviction that it will. This book contains my effort to provide a basis for its continuance.

LEONARD J. KOCH

ACKNOWLEDGMENTS

Some time ago, John Sackett of Argonne National Laboratory recognized that after 30 years of operations and staff turnover, current program personnel were not well-versed in the history of the EBR-II. While preparing a series of seminars to the Argonne-West staff to respond to this situation, I found that, although much of the historical information existed, it was not readily available or usable. Descriptions of what was done were retrievable but information detailing why was essentially unavailable.

It quickly became obvious to me that the record of EBR-II was incomplete and the story of a great adventure had not been told. In response to John's urging, I volunteered to lead an effort to get this story told before it was lost. Time was passing, and the potential story tellers were becoming unavailable. After some discussion, it was decided that I would write only the early history of the EBR-II project (and others would complete the record later).

Even this more modest undertaking proved to be quite challenging. It might not have succeeded without the support, encouragement, and assistance of Argonne National Laboratory staff, particularly Leon Walters. I am also deeply indebted to the following people:

- Carey Walton and Gail Walters for direct assistance in locating historical material and draft preparation
- Wally Simmons, Ralph Seidensticher, and Ron King for their reviews of the draft copy of the book and their advice and constructive comments
- Members of the staff at Argonne National Laboratory-West for a variety of supporting activities
- Members of the staff at RED, Inc. Communications for editing and page layout support.

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LEONARD J. KOCH

Chapter 1 — Introduction and Overview

Background

This book describes the history of the EBR-II during its development, design, construction, and initial operation. The operating history of the total power plant and fuel cycle, including its shutdown and closure are planned to be recorded in other documents. These activities occurred over a period of more than 40 years. During that period, changes in the technical and social/political environment influenced this history.

Early on, it was recognized technically that fast neutron reactors (unmoderated reactors) could utilize natural uranium and thorium most efficiently and that fast breeder reactors could utilize virtually all of the energy contained in uranium and thorium. Although the technical potential for high performance of the thorium–uranium-233 fuel cycle in fast reactors has been established, it has not been pursued actively in the United States and is not discussed further here.

Informally and unofficially there was general recognition and acceptance in the technical community that thermal reactors, primarily water cooled and gas cooled, were logical choices for near-term electric power generation; but for the long term, more efficient use of uranium would be necessary. It was in this setting that the EBR-II nuclear power system evolved and the overall concept of a fuel cycle and a power cycle for fast power reactors was developed.

This book also describes how the overall concept was developed to ensure that the knowledge, experience, and technology produced by EBR-II will be preserved and will be available to apply when this energy resource is needed.

Argonne National Laboratory

It would be incomplete to review the history of the EBR-II without also reviewing the organizational entity within which its development took place. The Argonne staff and management provided academic curiosity, superb scientific and technical capability, and a legacy of hands-on demonstrations of its ideas and concepts. The EBR-II concept evolved and materialized in this environment.

In the 1940s, there was general recognition by Argonne staff and management that atomic energy had tremendous potential, that its technical feasibility should be established, and the technology should be developed and demonstrated. EBR-I verified the theory of breeding, and the feasibility of operation of a liquid metal cooled fast reactor. This experience provided substance to the analyses that predicted the virtually unlimited potential of nuclear power. In late 1948, Enrico Fermi presented a seminar at Argonne in which he estimated the probable reserves of uranium and thorium in the world and converted the energy they contained to electric power. He concluded that these reserves could easily satisfy the United States electric power demand for several hundred years. Similar analyses by others produced similar conclusions.

During this period, the U.S. Atomic Energy Commission and the Joint Committee on Atomic Energy of the Congress were supportive of nuclear power development. The U.S. Atomic Energy Commission sponsored and supported (financially and technically) the development of various power reactor concepts. Argonne was involved in three programs:

- Technical support of pressurized water reactor development for the U.S. Naval Reactor Program
- Primary responsibility for developing and demonstrating the boiling water reactor concept by development, design, construction, and operation of the experimental boiling water reactor
- Primary responsibility for developing and demonstrating the liquid metal fast breeder reactor concept by development, design, construction, and operation of the EBR-II.

Although these programs proceeded essentially in parallel, there was general recognition of relative priorities. The U.S. Naval Reactor Program projected a near-term urgency. The boiling water reactor program had the objective of quickly advancing the basic technology of light water reactors to support early commercialization of nuclear power. While, the liquid metal fast breeder reactor program required long-range development to establish the ultimate capability of nuclear power. This judgement was verified by the relatively few years of operation of the experimental boiling water reactor, while EBR-II operated for more than 30 years (and additional operation could have advanced the technology even further).

EBR-II Organizational Structure

A unique aspect of this Laboratory activity involved the organizational structure which produced the EBR-II facility and its operation. There were multiple organizational structures, they overlapped, they interacted, there were multiple lines of authority, there were even divided responsibilities.

There were multiple organizational structures involved in EBR-II because development (including research and invention), engineering, design, and construction proceeded concurrently. Complicating matters further, oversight by the government was provided by three organizational units of the U.S. Atomic Energy Commission—Headquarters in Washington, DC, the Chicago Office of the U.S. Atomic Energy Commission, and the Idaho Office of the U.S. Atomic Energy Commission. Although the Chicago Office had oversight responsibility for engineering and construction, the Idaho Office administered the construction contracts.

The design and construction of EBR-II was accomplished by a temporary project organization superimposed on the permanent disciplinary organizational structure of the Laboratory where the supporting research and development proceeded concurrently. Many of the people in the project organization were also involved in the supporting research and development work. They participated in the development of the EBR-II technology and applied it to the design of the EBR-II facility.

As shown in Figure 1-1, the project organization was a relatively typical organizational structure reflecting the direct line project activities involved. The permanent organizational structure of the Laboratory (Figure 1-2) was directly and indirectly involved in the research and technical development support of the EBR-II project. This ongoing organization produced the technology which was incorporated into the EBR-II. This process involved continuing coordination and communication prompted by a common interest.

Contrary to established management concepts, this total organizational structure succeeded because the lines of communication were extremely effective. The project was faced with the usual requirements related to schedule and cost control while simultaneously depending on the development of the required supporting technology. The technology and concept were finalized concurrently by the normal compromises that must be made to achieve a conclusion. Some compromises resulted from a decision-making process where more than one option was available and which permitted a preferred choice. Examples of this process include the decision to locate the disassembly cell in the Fuel Cycle Facility rather than in the Reactor Plant. Another example was the use of mechanical pumps in the primary sodium system rather than direct current electromagnetic pumps (provisions were made to permit the substitution of electromagnetic pumps if necessary). On the other hand, some compromises were made because the technology was unavailable. For example, the EBR-II superheater concept could not be constructed because of the inability to make the unique tube-to-sodium tube sheet weld on the smaller diameter tubes.

A large number of people participated in both the EBR-II development program and the EBR-II project. Some on a part-time basis and some on a full-time basis. Some were involved primarily in design, incorporating the technology available or being developed, but most were involved in both because so much of the work required development and concurrent application of technology. This is typical of the development of a first of a kind product involving the application of very complex new technology.

The participants in the EBR-II project, including the development of the technology and the achievement of the EBR-II, are listed in Figure 1-3. It was not feasible to differentiate their participation because so much of it was dual. A separation was made between Fuel Cycle and Power Cycle, but even here there was some duality involving the fuel design and some aspects of fuel handling. Since it was not necessary to separate these activities to develop and accomplish the EBR-II, it is certainly not necessary to do so here. It is, however, essential to recognize the contributions and involvement of all of these participants. Finally, in the collegial

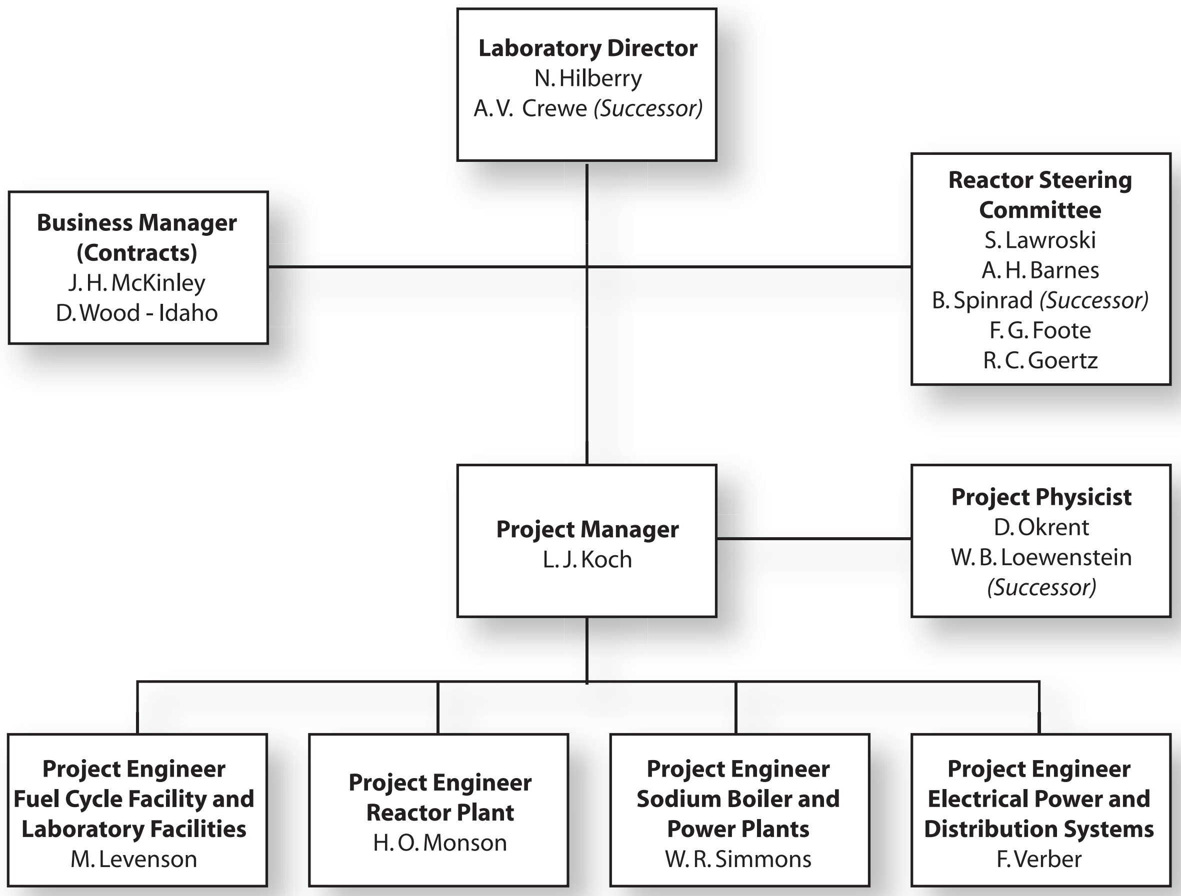


FIGURE 1-1. EBR-II PROJECT ORGANIZATION.

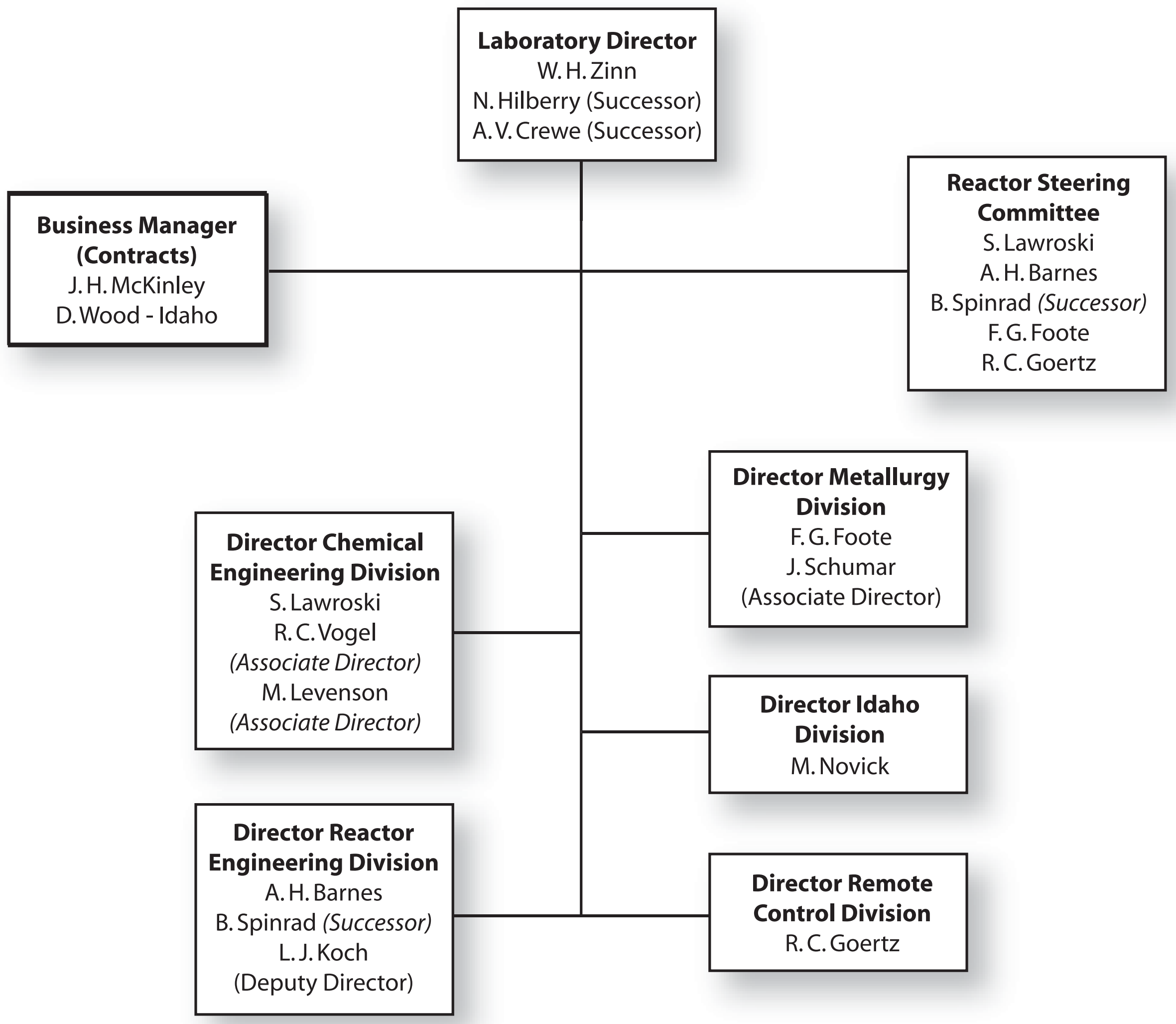


FIGURE 1-2. ARGONNE NATIONAL LABORATORY/EBR-II TECHNICAL SUPPORT ORGANIZATION.

FUEL CYCLE

M. Levenson
A. B. Shuck
J. H. Schraidt
J. E. A. Graae
K. R. Ferguson

POWER CYCLE

H. O. Monson
W. R. Simmons
F. Verber
E. Hutter
H. H. Hooker
K. D. Kuczen
G. K. Whitham
W. B. Loewenstein

J. E. Ayer	N. C. McCown	T. R. Bump	D. Okrent
G. A. Bennett	W. J. McGonnagle	J. P. Burelbach	J. A. Pardini
L. Burris	W. E. Miller	L. Burris, Jr.	C. A. Pesce
T. C. Cameron	J. H. Monaweck	J. D. Carter	R. R. Prina
N. J. Carson	M. V. Nevitt	B. C. Cerutti	R. L. Ramp
A. A. Chilenskas	R. A. Noland	S. H. Fistedis	M. B. Rodin
L. F. Coleman	R. H. Olp	J. D. Geier	A. D. Rossin
I. Dillon	E. J. Petkus	J. R. Ginder	V. Rutkauskas
R. J. Dunworth	R. D. Pierce	H. A. Hummel	H. Sandmeier
T. W. Eckels	G. Pokorny	J. R. Humphreys	C. H. Scheibelhut
H. M. Feder	N. A. Rodger	A. R. Jamrog	R. W. Seidensticker
M. J. Feldman	J. P. Simon	V. C. Jankus	O. S. Seim
N. R. Grant	E. S. Sowa	R. A. Jaross	F. A. Smith
A. P. Grumwald	C. E. Stevenson	W. J. Kann	E. S. Sowa
D. C. Hampson	R. K. Steunenberg	J. Kaeuis	T. R. Spalding
J. C. Hesson	C. Stone	H. Kasinskus	C. Stone
W. F. Holcomb	F. J. Tebo	W. C. Lipinski	T. E. Sullivan
H. F. Jelinek	D. E. Walker	A. Lovoff	F. J. Tebo
J. H. Kittel	J. R. White	D. E. Lutz	D. E. Walker
J. P. LaPlante	F. L. Yaggee	A. E. McArthy	
J. Ludlow	S. T. Zegler	R. A. Noland	

FIGURE 1-3. EBR-II PARTICIPANTS.

environment of Argonne National Laboratory there was strong interest and technical support from people totally uninvolved with EBR-II.

The Laboratory was also responsible for operation of the EBR-II facility. This responsibility was assigned to the Idaho Division which was responsible for all Argonne National Laboratory activities at the National Reactor Testing Station in Idaho, including operation of all Argonne National Laboratory reactor facilities. The Idaho Division was the equivalent of the commercial power plant owner/operator and accepted the transfer of the EBR-II facility from the EBR-II project. This transfer differed from a typical commercial transaction involving a nuclear power facility, because the Laboratory retained responsibility for EBR-II throughout the total life of the project. However, a formal written transfer of technical responsibility for the EBR-II activity from the project organization to the Idaho Division was implemented by the Laboratory Director. At that time the EBR-II project organization was dissolved.

The Idaho Division interacted with the EBR-II project organization in a similar manner as would an owner/operator with the Nuclear Steam Supply System engineer/builder of such a facility. However, since all of the participants involved in the process were employees of the Laboratory, this interaction was relatively informal and extremely effective. The future operators were very interested in, and concerned with design features, particularly those which related to operability and maintainability of the facility. Their comments and recommendations were an important component of the design process.

The Idaho Division had primary responsibility for preparation for operation. This included preparation of operating and maintenance manuals, procedures, and training manuals, and preparing operations and maintenance personnel, including their training and qualification. Needless to say, this complex interaction required effective communication and cooperation. EBR-II was a first-of-a-kind unit and all of these activities required coordination. The project organization was responsible for the design, but utilized input from the operators; while the Idaho Division was responsible for operations, but utilized input from the designers to prepare for operation of this first-of-a-kind facility. Designers described how they expected their equipment to operate and to be operated. The operators identified improvements to design which would enhance operations. This process was accomplished both formally and informally. The formal process included review of, and comments on, specific features and details. The informal process consisted of direct discussion and coordination between the interested parties and was by far the most used process. It was possible and effective because there were no barriers between the participants, they were all employees of the Laboratory and had a single goal. Although the design did not require the formal approval of the operator, the designers had every incentive to produce a design which the operator liked. The primary incentive was to develop a facility in which all parties were comfortable since that would most likely produce the desired product.

The transition from project to operations was smooth and efficient. The coordination and cooperation was most evident and most intense near the time of turnover. Prior to turnover, there were many activities that involved both organizations. Of particular significance was the start-up and testing of components, subsystems, and systems. These were normally performed by teams consisting of members of both organizations with participation and primary responsibility reflecting the personnel requirements for the specific activity being performed. Those activities which involved operations-type activities were led by operations staff and operating personnel with assistance from project staff. These included filling the primary and secondary systems with sodium, installing subassemblies in the reactor, and conducting critical experiments. Those activities that involved the start-up of a component, where the primary purpose was to ascertain that the functional requirements were fulfilled, were led by the responsible design engineer with assistance from operations staff and/or operating personnel. These included pumps, control drives, and fuel handling components.

This capability and opportunity undoubtedly contributed to the very successful operation of EBR-II. It did, however, have one negative impact. Because so much of this work was accomplished relatively informally, it did not produce an extensive formal record. This loss has made it more difficult to reconstruct a detailed record of the history of EBR-II, 30 to 40 years after these significant events occurred.

EBR-II Fuel Cycle

The EBR-II concept evolved around the fuel cycle. The primary objective of the overall concept was to achieve the fuel utilization, made possible by the breeding process. This approach to the development of a practical breeder reactor system produced the following basic objectives:

- The use of a high density fuel which minimized critical mass and enhanced the breeding characteristics of the reactor.
- The use of a fuel reprocessing system which recycled the fuel efficiently and quickly to minimize the total fuel inventory in the system.
- Demonstrate the feasibility of achieving a total plant operating cycle which required only the addition of uranium-238 to sustain plant operation.

The Metallurgy Division was encouraged in their development of metallic fuel because it provides the highest fuel density of the many possible fuel compositions, and their compatibility with sodium, the preferred reactor coolant was verified by EBR-I. The primary disadvantage of uranium metal fuel is its susceptibility to irradiation damage. However, early work with uranium metal alloys indicated that the irradiation damage resistance might be enhanced by the addition of small amounts of alloying materials.

At the same time, the Chemical Engineering Division was investigating pyrometallurgical processes for removing fission products from irradiated nuclear fuel. These processes had the advantage of being very compact and of recovering the fission products in a very concentrated form. They had the disadvantage of not removing all of the fission products, and therefore the processed fuel was highly radioactive. However, the primary fission product contaminants were noble metals (primarily ruthenium and molybdenum) which had the potential to act as stabilizing alloying agents in the uranium metallic fuel alloy.

These processes for enriched uranium metal fuel alloys were reasonably well-developed and understood. It appeared that they would be feasible for the initial operation of EBR-II. Further, development of pyroprocesses for recycle of plutonium-uranium metal alloys appeared promising, but required additional detailed development of processes and equipment. Of perhaps greatest significance was the fact that it appeared that the same basic facilities could be used, with different process equipment, to apply and demonstrate integrated fuel cycles with both fuel systems (i.e., enriched uranium metallic fuel alloy and plutonium-uranium metallic fuel alloy). Although there were many unresolved problems and uncertainties, there appeared to be a technical fit. The purpose of EBR-II was to make this a reality.

The EBR-II reactor and fuel cycle were developed on the basis of initial use of enriched uranium fuel alloy in the reactor and fuel cycle with the expectation of switching to a plutonium-uranium fuel alloy at a later date. This program was intended eventually to achieve the ultimate objective and demonstrate the integrated operation of power cycle and fuel cycle utilizing a plutonium-uranium-238 fuel cycle. The basic feasibility of manufacturing and assembling fuel assemblies using highly radioactive fuel materials needed to be established. These operations involved complex fabrication, manufacturing, and assembly procedures performed by remote control in heavily shielded facilities that required demonstration.

EBR-II may be unique in the development of nuclear power plants with respect to the influence of the fuel cycle on the overall plant design and operation, but it reflects the need to integrate the fuel cycle into the total operation for nuclear power plants operating on recycled fuel.

EBR-II Power Cycle

The required end product of this program was electricity. Therefore, one of the principle objectives of the EBR-II program was to demonstrate the reliable, efficient generation of electric power. Also, to the extent practicable with a small experimental unit, to demonstrate the delivery and availability under conditions comparable to those existing for commercial power generating stations. The EBR-II reactor and Power Plant were designed to achieve these objectives and did so for more than 30 years; even when the reactor was being operated as an experimental irradiation facility, the plant operated primarily as a base load power station.

However, a more fundamental requirement of the primary coolant system, beyond removal of the heat generated in the reactor for power production, was the removal of fission product decay heat after reactor shutdown. This is a requirement unique to nuclear power reactors and is proportional to the power density in the reactor when operating at power. Typically, fast neutron power reactors operate at high power density and shutdown cooling requirements are quite severe. The goal was established to achieve this passively, without the operation of active (powered) systems.

The primary technological emphasis on the EBR-II power cycle was directed to the reactor and primary sodium cooling system. It was recognized that very significant technical advancement from EBR-I and other early work would be imperative if EBR-II was to achieve its objective of advancing the technology to the interesting stage for future commercialization. Therefore, a relatively conservative approach was taken to the technology applied to the balance of plant for EBR-II. There was no effort to develop advanced power cycles or power equipment technology (i.e., the steam conditions selected were quite common for EBR-II size commercial fossil fueled units). On the contrary, the emphasis was placed on reliability and operability. It was thought that the EBR-II fuel cycle should demonstrate the potential for very low fuel cost. Thermal efficiency is less important in systems with low fuel cost; capital cost becomes more significant in establishing the cost of power produced. Also, since capacity factor impacts power generation economics, it was a desirable attribute for EBR-II to demonstrate high capacity factor.

EBR-II Initial Operation Objectives

Power operation of EBR-II began in August 1964 when electric power was first delivered to the National Reactor Testing Station distribution grid. This definitive phase of operation was preceded by a variety of preparatory operations and experiments.

This initial operation of the plant was in accordance with the basic intent and objective of operating a liquid metal cooled fast breeder reactor as an electric power generating plant operating on recycled fuel. Except for a relatively slow and extended start-up process of increasing power levels and examination of fuel (for evidence of irradiation damage), the start-up of EBR-II proceeded well. It operated as an experimental electric power generating station, delivering power to the National Reactor Testing Station 138 kilovolt power loop (about 15,000 kilowatt net power). After the start-up and approach to power phases were completed, the EBR-II plant operated for more than 30 years as a base load generating plant, most of it at its rated power of about 20 megawatt electric.

The initial phase of EBR-II operation lasted almost five years. During that time, the primary emphasis was on verifying and demonstrating the operating characteristics of the power cycle and the fuel cycle. Particular attention was given to the feasibility of the unique EBR-II fuel cycle, especially the processing and fabrication of highly radioactive fuel on site and the power performance of recycled fuel. These operations proceeded extremely well and much experience was gained in the process. The fuel was recycled through the reactor and Fuel Cycle Facility about four times; about 35,000 fuel elements and 400 fuel subassemblies were reprocessed and fabricated on site.

During this time, the U.S. Atomic Energy Commission revised the liquid metal cooled fast breeder reactor fuel development program and assigned essentially total support to the development of uranium-plutonium mixed oxide fuel for liquid metal cooled fast breeder reactors. This resulted in a major change in the EBR-II program. The reactor program was altered to accommodate irradiation experiments on mixed oxide fuels, and the Fuel Cycle Facility was converted to a “Hot Cell” for examination of irradiation experiments. However, because of its excellent performance, the reactor continued to be fueled with the basic enriched uranium/fissium alloy, but it was not recycled. Each fuel loading was fabricated from fresh, enriched uranium, alloyed to simulate recycled fuel, and the spent fuel was placed in storage. This 25-year supply of stored EBR-II spent fuel is included in the Department of Energy stockpile of recoverable material. Its recovery is a part of the EBR-II closure plan.

EBR-II was shut down on September 30, 1994, in accordance with an operational plan developed in response to requirements established by the Department of Energy. This plan incorporated provisions for maintaining the necessary operating systems in either an operational mode or standby mode, as appropriate.

CHAPTER 2 — DEVELOPMENT OF THE EBR-II CONCEPT

The EBR-II concept evolved slowly and along several parallel paths. With limited technology available — EBR-I was the state of the art at the time — developing a working concept required innovation and invention.

The EBR-II concept was strongly influenced by the limited availability of highly enriched fuel at that time. Military applications had priority for using highly enriched uranium and plutonium. It appeared that these materials would have limited availability and would be costly in the future. Consequently, the EBR-II concept was based on the need to achieve a high power density in the reactor and minimize the total fuel inventory by all means available.

EBR-II FUEL ELEMENT

It was evident that an entirely different fuel system was needed. To achieve a much higher power density required a different geometry. To maintain reasonable fuel temperatures, a smaller diameter fuel element would be needed. A smaller fuel element led to a fuel assembly concept that made the fuel package, "the fuel subassembly," the cornerstone of the power cycle.

A tentative target thermal power density of about 1 megawatt per liter of core volume was established for design purposes; the EBR-I operated with a power density of less that 1/6 of a megawatt per liter.

This target power density provided a basis for establishing the physical and thermal parameters that define the reactor core. Although sodium has excellent thermal conductivity, it has relatively low specific heat and requires the movement of a relatively large volume of coolant to remove the heat. These considerations resulted in a configuration that produced high heat flux from the fuel to the coolant, which in turn required a small cross section for the fuel element to maintain acceptable fuel temperature and a large cross section for coolant to provide the necessary flow volume to remove the heat.

Another parameter for a realistic design was coolant temperature rise through the reactor of about 200°F and maximum coolant flow velocity of about 25 feet per second. These reactor conditions coupled with a realistic and conservative steam condition of 850°F and 1,250 pounds per square inch, resulted in reactor operating conditions of about 700°F sodium inlet temperature and 900°F sodium outlet temperature.

Early on, a closely packed hexagonal geometry was selected for the EBR-II reactor configuration. EBR-I had demonstrated improved stability by incorporating cylindrical fuel elements in hexagonal subassemblies as a refinement and modification of the original design.

A series of iterations resulted in a 0.174-inch diameter fuel element and a pin diameter of 0.144 inches for the uranium alloy pin. Since the EBR-II concept was predicated on the use of recycled fuel from highly radioactive uranium alloy that was fabricated remotely, the feasibility of fabricating such small diameter fuel pins had to be established. The usual processes were totally impracticable in the high radiation, remote-controlled environment in which the fabrication would take place. The problem was resolved by the development of a casting process that produced a finished precision metal casting of the proper diameter that could simply be cut to the proper length.

Without this development it was quite likely that the EBR-II fuel recycle concept as conceived at that time would have been unsuccessful. However, subsequent operation of EBR-II demonstrated that the EBR-II fuel element design was overly conservative, and that the desired thermal performance could be achieved with a larger diameter fuel element, approximately 1/4 inch in diameter. This increased diameter would have made all of the fabrication steps much easier, and was being considered for the plutonium-uranium fuel cycle for EBR-II.

EBR-II SUBASSEMBLY

The small size of the individual EBR-II fuel elements required that they be handled and loaded into the reactor as a group or package. That package was the fuel subassembly and consisted of a hexagonal stainless steel tube containing:

- A cluster of 91 fuel elements
- An upper and lower blanket region
- A lower adapter for positioning and support in the reactor and elsewhere in the fuel cycle
- An upper adapter for attachment to various devices utilized to handle, transfer, and transport the subassemblies.

The EBR-II fuel subassembly is shown in Figure 2-1.

The freestanding subassembly concept represented a major departure from EBR-I, which incorporated a series of grids to support the fuel elements. The EBR-II reactor concept consisted of three different types of free standing subassemblies, positioned and supported only by a bottom grid-plenum structure.

The purpose of the subassemblies was to permit assembly of the proper amount of fuel, blanket, and structure in the proper configuration to constitute a nuclear reactor. Individual subassemblies containing properly configured materials were loaded into the reactor in a prescribed sequence.

Initially the reactor was manually loaded with 637 non-nuclear, stainless steel dummy subassemblies. After that, reactor loading was accomplished by removing one subassembly and replacing it with another to reach the desired loading. Similarly, after reactor operation was terminated in 1994, the reactor was defueled on the same replacement basis.

The lower grid plate of the reactor established the configuration of the reactor. Three patterns of different sized holes in the reactor grid matched the lower adapters of the core, inner blanket, and outer blanket subassemblies and produced the configuration shown in Figure 2-2. This figure also shows locations of control and safety rods that were smaller in cross-section by the equivalent of one row of fuel elements. The thimbles, or hexagonal tubes, in which the control and safety rods move vertically, have the same external hexagonal dimensions as the fuel and blanket subassemblies.

Two configuration features were incorporated into the EBR-II concept as demonstrations for potential application in large future liquid metal cooled fast breeder reactors. The first was a provision to prevent installing a subassembly into an incorrect reactor location. The fuel

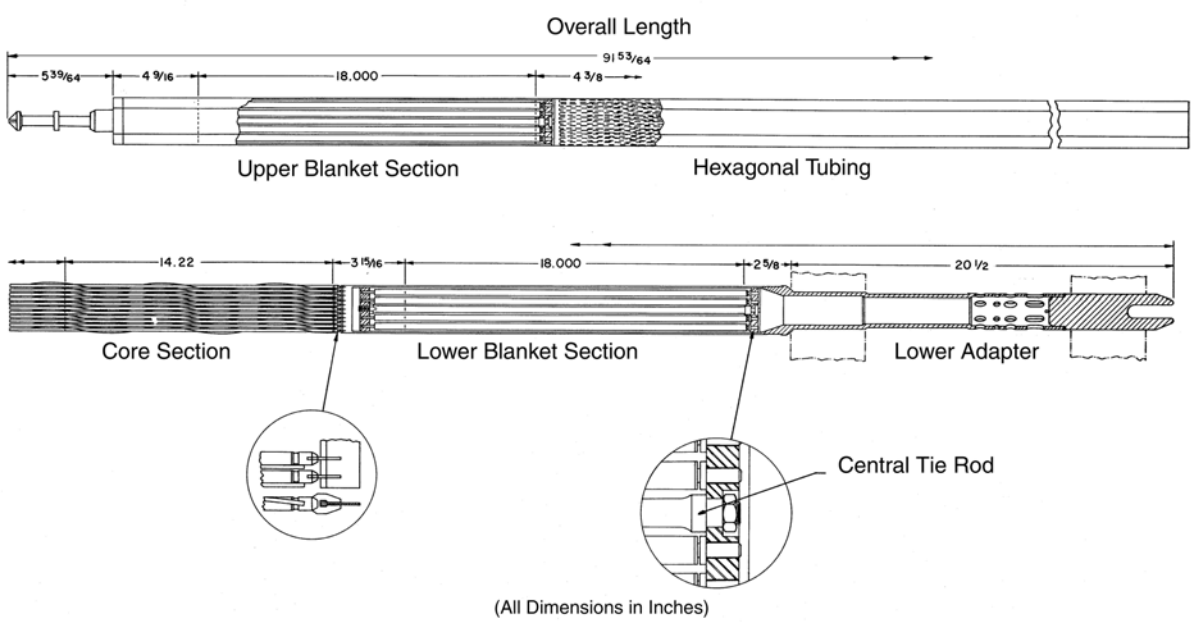
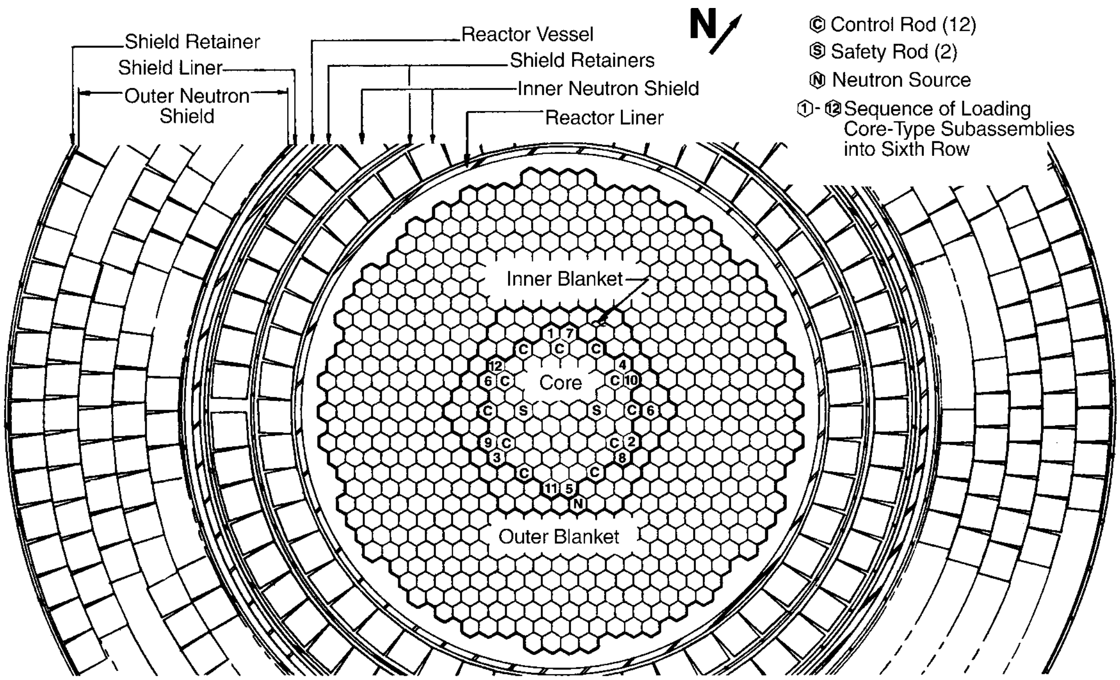


FIGURE 2-1. CORE SUBASSEMBLY (FINAL CONFIGURATION).



Note:
For nominal (67 subassembly) core, Nos. 1 through 6 are only loaded with core type, Nos. 7 through 12 with inner blanket type.

FIGURE 2-2. REACTOR ARRANGEMENT.

subassemblies had a lower adapter with a larger diameter than the other subassemblies and could not be installed in either of the two blanket locations. Similarly, the inner blanket subassemblies could not be installed in the outer blanket. In the opposite direction subassemblies could not be installed closer to the reactor center than their proper zone because the orientation bars, which engaged slots in the bottom of the lower adapters, become wider toward the center of the reactor.

The second configuration feature involved local coolant flow through the subassemblies to control coolant outlet temperature. The subassembly power density decreased as radial distance increased from the center of the reactor. The objective was to demonstrate that controlled coolant flow could be accomplished without placing individual flow control orifices in each subassembly.

The initial approach was to place slots (Figure 2-3) in the lower adapter of each subassembly and to provide steps in the lower grid plate of the inlet plenum to cover a part of the slot, and thus affect flow through slots and into the subassembly as shown in Figure 2-4. This did not work, but the concept was modified to place holes in the adapter rather than slots, as shown in Figure 2-1.

By making each subassembly type identical with identical inlet coolant holes at different elevations and providing steps in the lower grid plate for each row, the amount of coolant flow could be adjusted to correspond to the power generation in that row of subassemblies. This concept was applied to the five rows that constitute the reactor core and the two rows that constitute the inner blanket shown in Figure 2-2. Although this concept was of limited value in a small reactor such as EBR-II, it could be extremely useful in a large reactor and was incorporated and demonstrated for that reason.

Because of the wide variation in power density between the fuel subassemblies and blanket subassemblies, the sodium coolant system was divided into a high pressure system for the core and inner blanket and a low pressure system for the outer blanket.

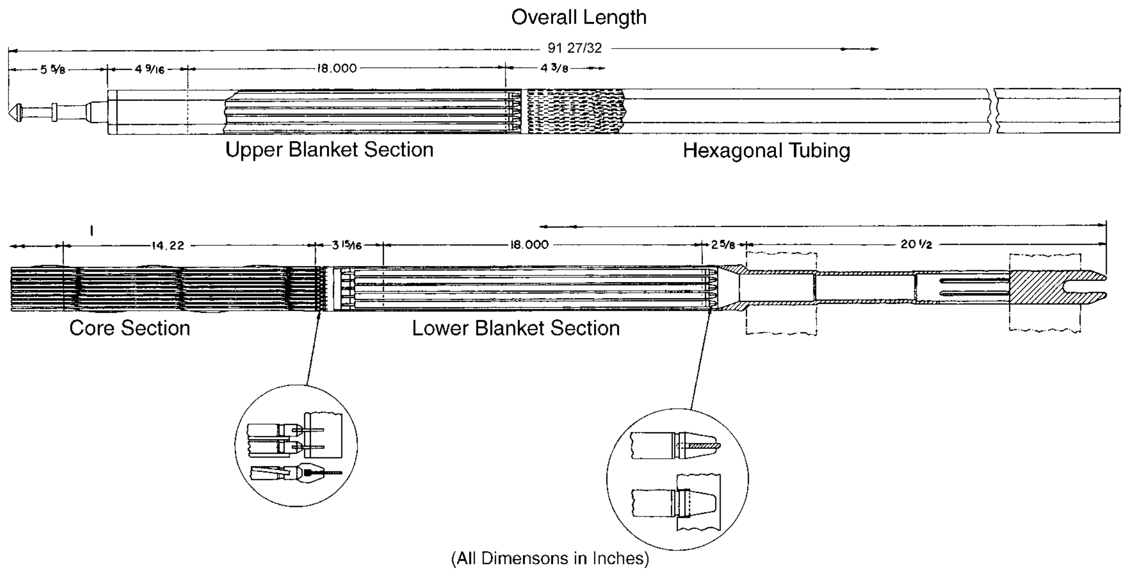


FIGURE 2-3. EBR-II CORE SUBASSEMBLY (EARLY DESIGN).

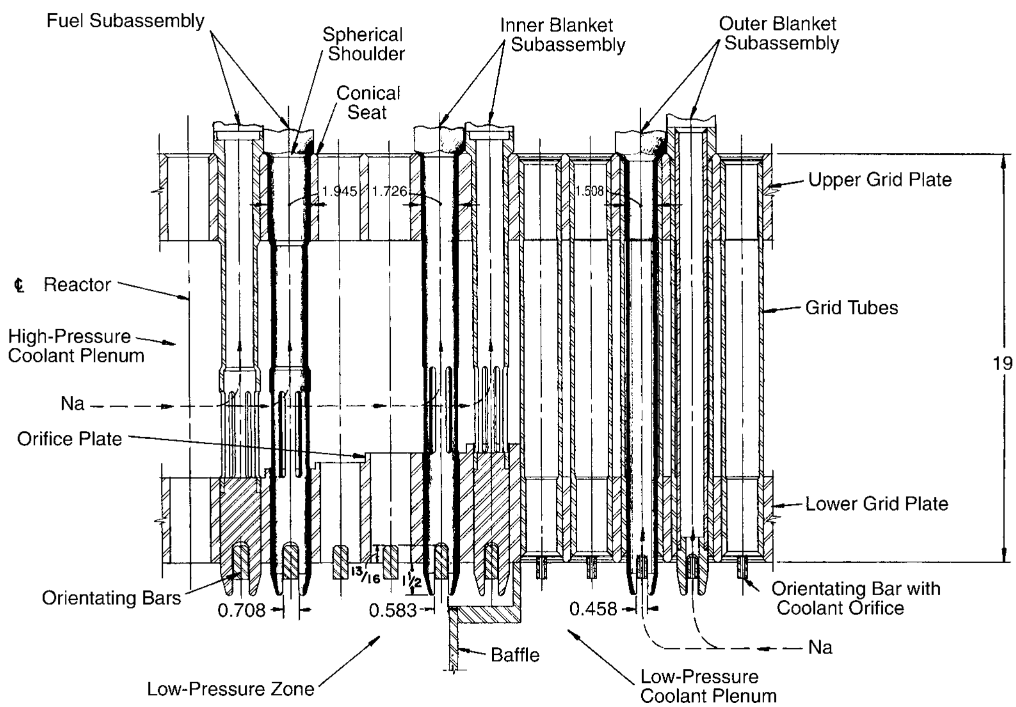


FIGURE 2-4. REACTOR SUPPORT GRID (EARLY DESIGN).

The pressure drop of the coolant flowing through fuel subassemblies was significant enough to lift the subassemblies. This was unacceptable during operation. Mechanical provisions to prevent such lifting could have been incorporated into the design, but there was a strong incentive to avoid incorporating any latches or locks in the assemblies. Stops above the subassemblies to prevent lifting would have had to accommodate thermal expansion of the subassemblies.

A non-mechanical solution was incorporated to permit the subassemblies to expand. The high pressure inlet sodium coolant plenum and the subassemblies were arranged to provide downward hydraulic pressure on the subassembly to offset the upward lifting force of the coolant flow. By introducing the inlet flow into the interior of the lower adapter, pressure was imposed on the closed bottom of the adapter. The hydraulic hold-down force, plus the weight of the subassembly exceeded the lifting force, and no other provisions for hold-down were needed.

EBR-I had demonstrated that mechanical bowing of fuel elements toward the center of the reactor caused by the power gradient across the reactor produced a positive power coefficient. In EBR-I the power density decreased across the diameter of the cylindrical fuel elements depending on their radial position from the center of the reactor.

EBR-II presented a greater potential for a positive bowing power coefficient. In EBR-II the bowing would be produced by a temperature gradient across the fuel subassembly hexagonal tube while the EBR-I temperature gradient was across the much smaller diameter fuel element tube. The EBR-I experience verified that the observed positive power coefficient was produced by a thermal mechanical effect and not a nuclear characteristic. Therefore, an EBR-II concept imperative was the requirement that thermal effects would not produce physical change that would result in a positive power coefficient.

The EBR-II physical and structural configuration was established by positioning and supporting the subassemblies in the grid-plenum structure that was unaffected by the power level of the reactor. This grid-plenum structure was at the temperature of the inlet sodium, which was at the temperature of the bulk primary sodium. Therefore, the thermal and physical effects had to be controlled above the fixed support structure. In response, the reactor was designed to enhance favorable thermal expansion and ensure that bowing would be prevented, or limited to an acceptable level.

The subassemblies of the EBR-II had to be replaced periodically, and therefore had to be movable. Any clearance for movement had to meet the requirements established to prevent subassembly bowing. Appropriate local clearance that satisfied the bowing requirement was achieved by incorporating a button on each of the six sides of the hexagonal subassembly tubes at approximately the vertical center of the reactor (Figure 2-5).

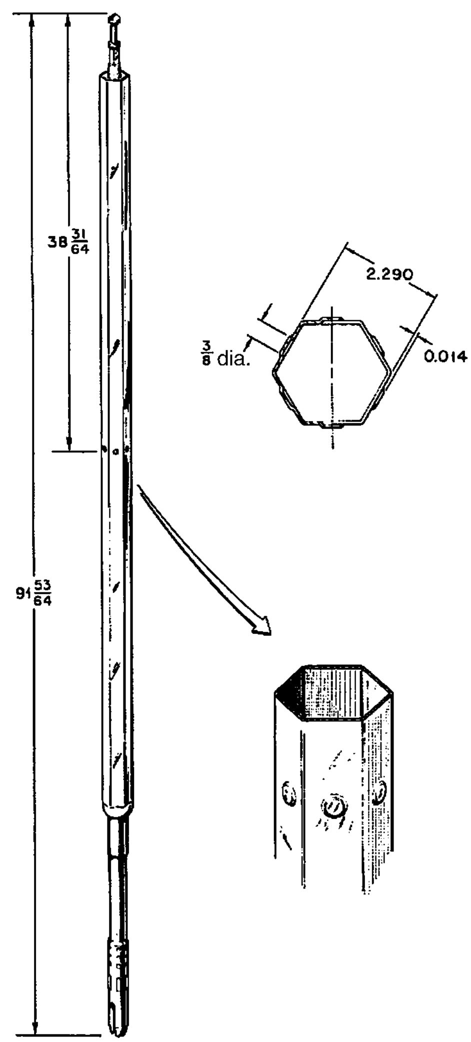


FIGURE 2-5. EBR-II SUBASSEMBLY-SPACER BUTTON DETAILS.

The nominal clearance between the buttons with the subassemblies in place was 0.002 inches. This small clearance produced a very tight structural configuration at the vertical mid-plane of the reactor. The subassemblies attempted to bow toward the center of the reactor, but because they were anchored at the bottom in the grid structure and were confined at the midpoint by the buttons, they tended to bow outward above the buttons.

This solution produced a favorable component of the power coefficient. It should be noted, however, that the unrestrained axial thermal expansion of the upper end of the subassemblies was made possible by the hydraulic holddown concept incorporated into the EBR-II design.

EBR-II Reactor Control

Leakage control was not a feasible option for EBR-II. The use of neutron absorbers was questionable because of nuclear performance uncertainties, but also because of the desire to demonstrate high neutron efficiency. Maximizing breeding ratio was an objective of EBR-II. The movement of fuel appeared to be feasible and was compatible with the basic EBR-II concept of reactor and subassembly. It was recognized that in the EBR-II reactor configuration, a guide would be required for any moveable unit in the reactor, which naturally led to the hexagonal thimble concept located by, and supported in the same manner, as all of the subassemblies. A fueled control rod which would fit in such a guide was made smaller than a fuel subassembly by one fewer rows of fuel elements. The reactor configuration would accommodate 12 such control rod and thimble assemblies.

All subassemblies were freestanding, supported only at the bottom in the reactor grid/plenum and were easily lifted during fuel handling. This arrangement for the control rod thimbles was not acceptable, since the vertical movement of the control rods could lift the thimbles. But it was imperative to incorporate the control rods into the basic EBR-II reactor concept of freestanding hexagonal-shaped containers for the required components comprising the reactor. To prevent movement of the control rods, the thimbles were locked into place by a latching arrangement at the lower end which was effected by rotating the thimble 60 degrees during installation (Figure 2-6). Since rotation of the thimble was prevented by the six adjacent hexagonal subassemblies, a special operational sequence was required to remove and replace a thimble. First, the six adjacent subassemblies were replaced by special dummy subassemblies, with the side adjacent to the thimble scalloped to permit the thimble to be rotated. An attachment had to be made to the upper end of the thimble, which was an open hexagonal tube, that simulated the upper adapter of a subassembly and provided the capability for the fuel handling and transfer machines to remove and replace the thimble. The thimble was replaced, after which the six dummy guide subassemblies were replaced by regular subassemblies. This was a tedious and time-consuming operation, but was required only rarely. Most importantly, it was accomplished without violating the basic requirements of the EBR-II fuel handling concept that there was never more than one vacant lattice position in the reactor at any time and that all components of the reactor consisted of removable, freestanding units. A similar procedure was used for the safety rod thimbles.

Subassembly as a Container of Fuel and its Transfer and Transport

The subassembly was a package in which the fuel elements could perform the function of generating heat while providing the physical capability for that heat to be removed and used productively. This function required the ability to install and remove the fuel subassemblies many times over the operating lifetime of the reactor. Because the EBR-II concept included fuel recycle, extraordinary and unique requirements were imposed on these activities.

In EBR-II, fuel handling consisted of removing and installing subassemblies in the reactor. The concept assumed that fuel handling operations could be required quite frequently, even as often as weekly. This aspect of the fuel handling concept was influenced primarily by uncertainty about the irradiation damage resistance of the fuel. But another influence was consideration of an operating strategy that might be favorable for commercial power generation — refueling the power reactor over a weekend when power demand was lower than during the workweek.

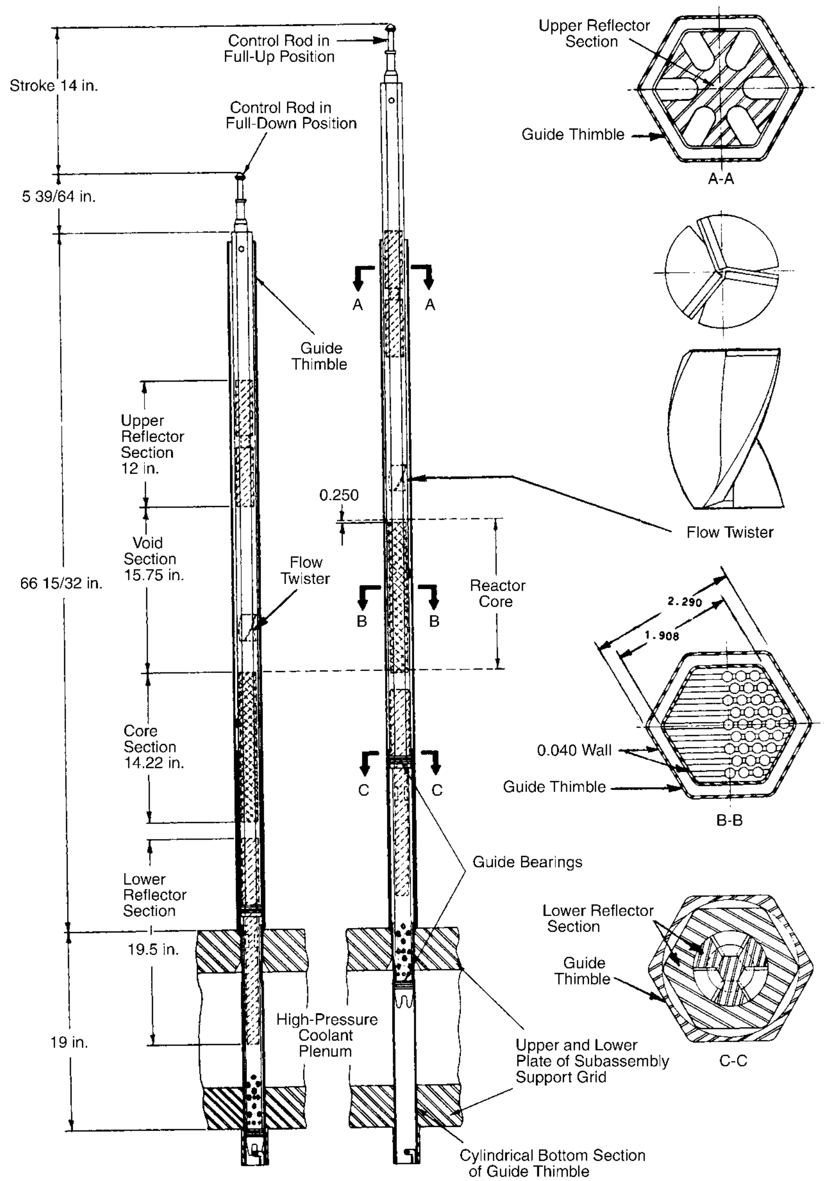


FIGURE 2-6. CONTROL SUBASSEMBLY.

The EBR-II fuel alloy proved to be very durable. After a series of design improvements, 10 percent fuel burnup was achieved routinely, significantly reducing the frequency of refueling. Nevertheless, the capability to perform very rapid fuel handling operations proved to be invaluable in supporting experimental programs.

The EBR-II fuel handling concept incorporated an intermediate storage capability in sodium because the subassembly could not be removed from the liquid sodium environment directly. A storage rack was provided in the primary tank. The operation of the reactor thus was made independent of the transfer and transport of subassemblies to the Fuel Cycle Facility. The time required to replace fuel in the reactor was minimized and passive heat removal was accomplished by natural convection of the sodium in which the subassembly remained submerged. Since the subassemblies continued to be cooled in the storage rack, reliable storage capability was provided indefinitely. This capability was consistent with the EBR-II reprocessing concept of cooling the fuel for as little as 15 days prior to reprocessing, or as long as desired.

The total integrated fuel transfer and transport concept between the Reactor Plant and the Fuel Cycle Facility involved much study and evaluation. Some of the key considerations involved how and when to make the transition from the sodium cooling environment and how to ensure continued reliable cooling of the spent fuel. Fission product decay heat removal occurred easily and reliably in sodium by natural convection, but it was clear that forced convection would be required in an inert gas environment. These cooling requirements would exist for the reprocessed fuel being returned to the reactor at all times that the fuel elements were clustered in the subassembly. Unclustered fuel elements would self cool.

The evolution of the fuel transfer/transport concept focused on the transition from sodium to inert gas coolant medium and the disassembly and assembly process involved in the transition of fuel elements between a tight configuration and a loose configuration. One of the early objectives in the evolution of the concept involved taking the subassembly apart quickly after removal from the sodium coolant in the primary tank.

To achieve this objective, a disassembly cell above the primary tank at the storage basket location was incorporated into the early design concept. In this arrangement the subassemblies were to be transferred from the storage rack directly to the disassembly cell and mechanically disassembled to remove the fuel elements as shown in Figure 2-7. (This concept was not used.)

Although this concept was retained well into the design phase, it was replaced by the final design because of concerns about possible impact on reactor operations and the advantages of physically separating the fuel cycle and power cycle operations.

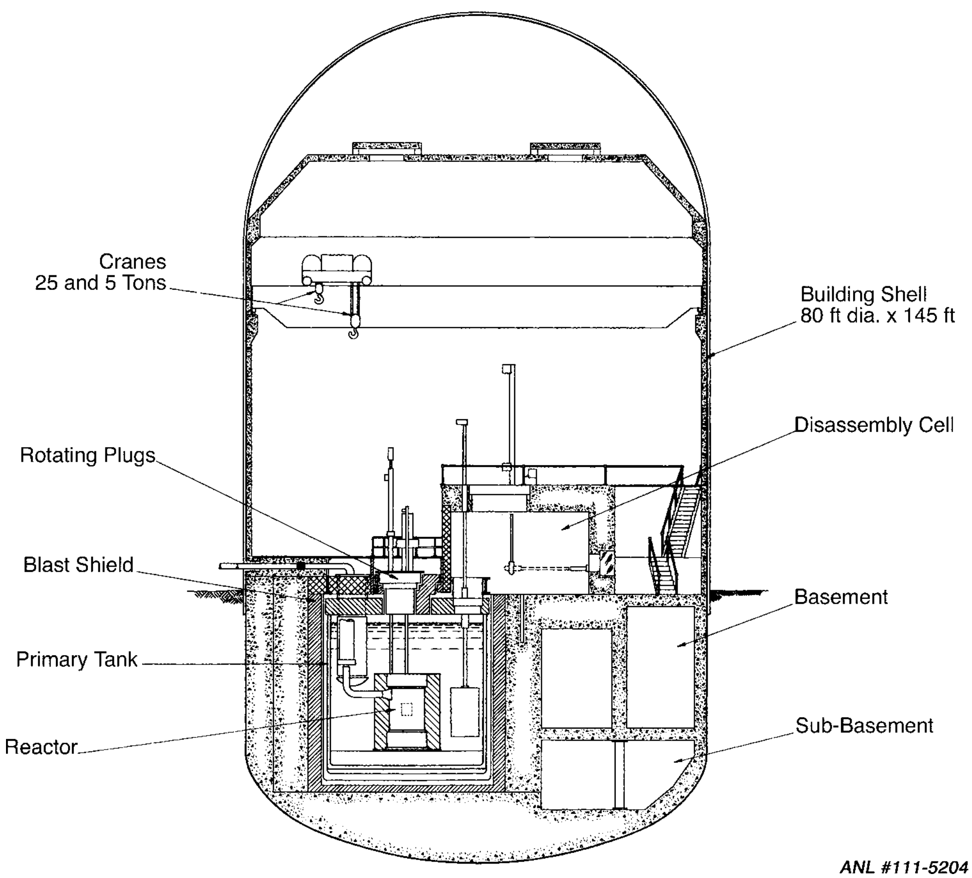


FIGURE 2-7. EBR-II REACTOR PLANT (VERY EARLY DESIGN).

An interesting consequence of this scenario was that the depressed floor area of the disassembly cell was retained and later provided valuable space for experimental systems and components. Figure 2-8 shows the design of the fuel transfer/transport concept after the disassembly cell was deleted.

Elimination of the disassembly cell resulted in the addition of the air cell in the Fuel Cycle Facility and the fuel unloading machine in the Reactor Plant. The cooling environment for the subassembly shifted from liquid sodium to argon gas as the subassembly was lifted into the fuel unloading machine. Forced circulation of argon gas was provided in the fuel unloading machine and in the inter-building coffin during transport and until the residual sodium had been washed from the subassembly components. At that point, heat removal was provided by forced circulation of air until the subassembly was opened and the fuel elements separated from the close packed tight hexagonal geometry. When separated, the fuel elements were cooled sufficiently by natural circulation of air. Because all of the fission products were not removed, decay heat removal was required for the reprocessed fuel. The same equipment and operations were used in the return of the reprocessed fuel to the subassembly storage rack in the primary sodium. Similarly, the same scheme was used subsequently when experimental subassemblies were returned to EBR-II after interim examination. The intermediate storage capability incorporated into the EBR-II concept made an integrated operation of two quite dissimilar operations possible and efficient.

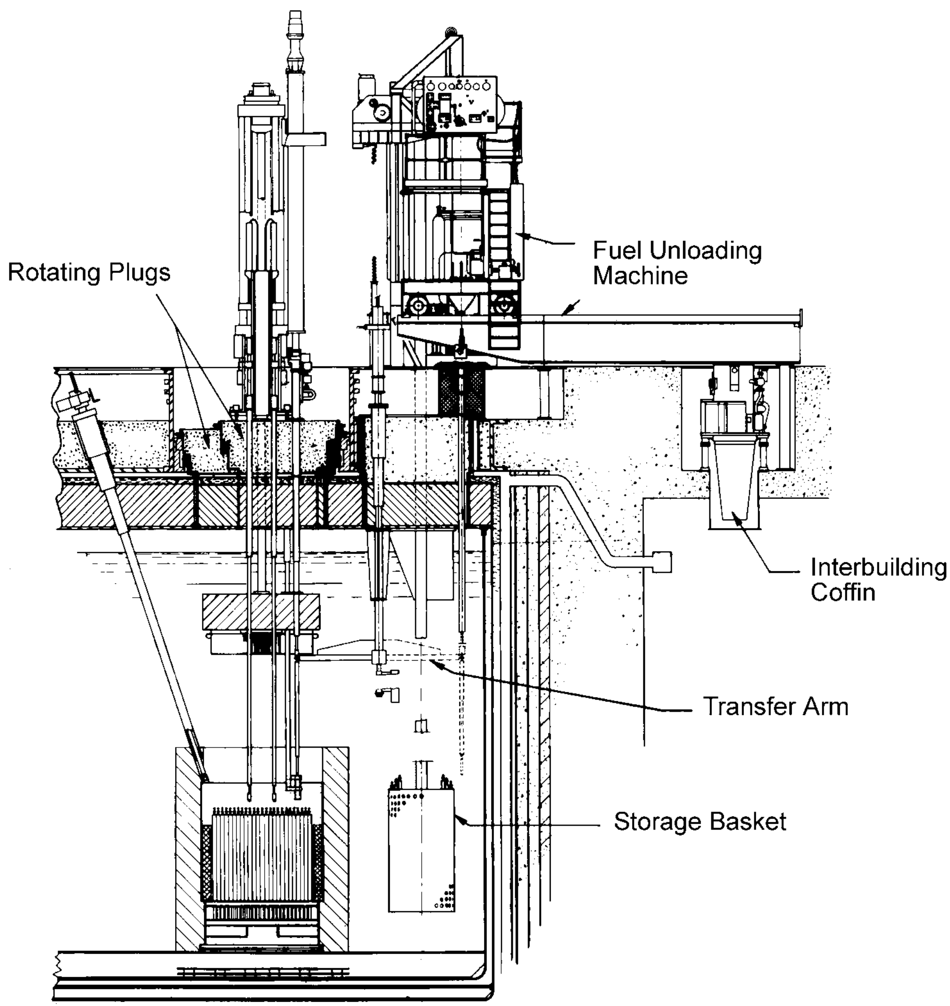


Figure 2-8. FUEL HANDLING SYSTEM WITHOUT DISASSEMBLY CELL.

The details of the design features of the various components comprising the total fuel handling, transfer and transport systems are described later, but the basic concept and its influence on the design can be summarized as follows:

- The same operational requirements, processes, and equipment would be capable of handling all of the subassemblies and related components that were installed in and removed from the reactor.

- Fission product decay heat removal would be ensured at all times during the process.

- At the appropriate times in the process, the subassembly would transfer from a sodium environment to a gas environment, with an attendant change in coolant.

- To reduce the impact of this transition in coolant medium, the EBR-II concept was based on 15 days minimum storage time in sodium coolant before transition to a gas coolant occurred.

- In the reverse scenario, when a subassembly was being delivered to the primary system, adequate preparation had to be made for the subassembly to accommodate immersion in 700°F sodium.

The EBR-II reactor concept introduced some unique requirements during fuel handling. The subassemblies were held in close-packed position by their weight and engagement in the grid. The subassembly involved in the fuel handling operation was lifted from the close-packed cluster of subassemblies. To address concern about the six subassemblies that surrounded the one being removed, a hold-down feature was added to the gripping and lifting sequence involved in removing the subassembly. The hold-down feature was augmented by a spreading feature to move the six surrounding subassemblies away from the one being removed (Figure 2-9).

Swelling of stainless steel and other distortions incurred during long-term residence in the reactor caused interference between some subassemblies. Extra force was required to remove these subassemblies and the hold-down spreading feature permitted the application of such force without jeopardizing the stability and reliability of the reactor configuration. Figure 2-10 is a photo of a removed subassembly that shows the result of interference with other subassemblies.

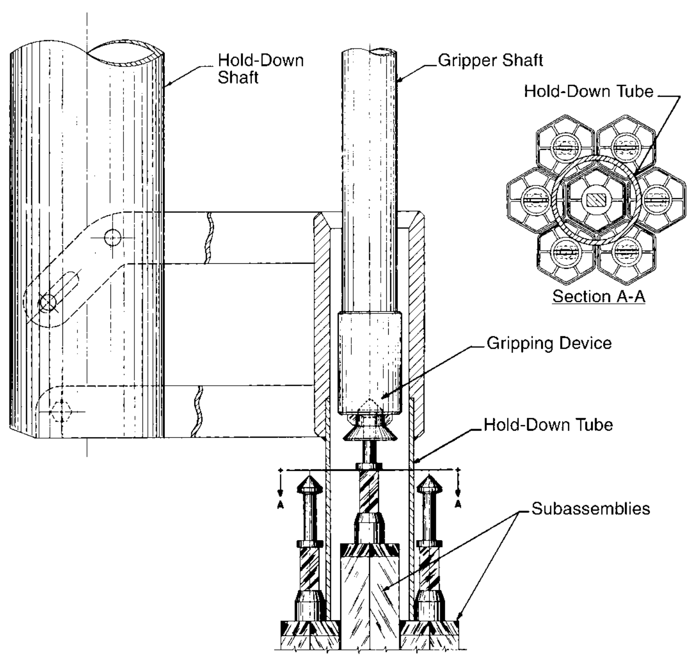


FIGURE 2-9. SUBASSEMBLY HOLD-DOWN AND GRIPPER.

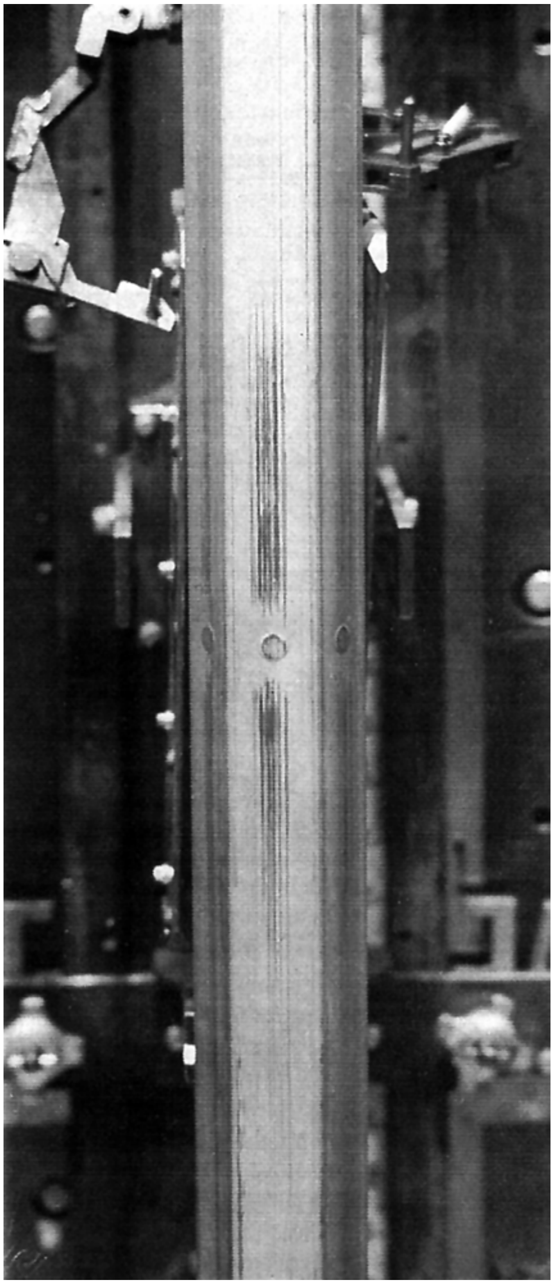


FIGURE 2-10. HIGH-BURNUP SUBASSEMBLY.

THE EBR-II REACTOR AND PRIMARY SYSTEM CONCEPT

In the EBR-II reactor concept, sodium coolant was provided to the reactor grid plenum by two mechanical centrifugal pumps from which it flowed through the subassemblies removing the heat generated by fission of uranium (or plutonium). The heated sodium flowed from the reactor to an intermediate heat exchanger where the heat was transferred to the secondary sodium system. This very simple flow system is shown in Figure 2-11. The primary sodium was radioactive because it flowed through the reactor and was exposed to neutrons. The secondary sodium was not radioactive.

The reactor and primary sodium system were contained in the primary tank and were completely submerged in sodium (see Figure 2-12). Principal benefits included:

- Fuel handling with intermediate storage
- Coolant system containment reliability
- A simple double-walled tank with no openings or penetrations below the sodium level
- Large capacity to absorb heat provided by the large volume of sodium at reactor inlet temperature.

This concept evolved as the additional, specific requirements for reliable heat removal were identified.

As these requirements were evaluated it became apparent that reliable fission product decay heat generation could be more demanding and critical than heat removal during power operation of the reactor. This characteristic resulted from the fact that heat removal during power operation was an on/off situation and could be turned off very easily. On the other hand, fission product decay heat generation continued irrespective of circumstances and could not be turned off. This heat had to be removed reliably at all times, under all conditions, and in all environments. The EBR-II concept required passive heat removal. The submerged primary system concept provided a direct and reliable capability to satisfy this requirement.

Refinements were developed to meet the various conditions that could exist after reactor shutdown. For example, conditions could exist in the secondary sodium and steam systems at shutdown that would affect the heat removal capability through the intermediate heat exchanger from the primary sodium system.

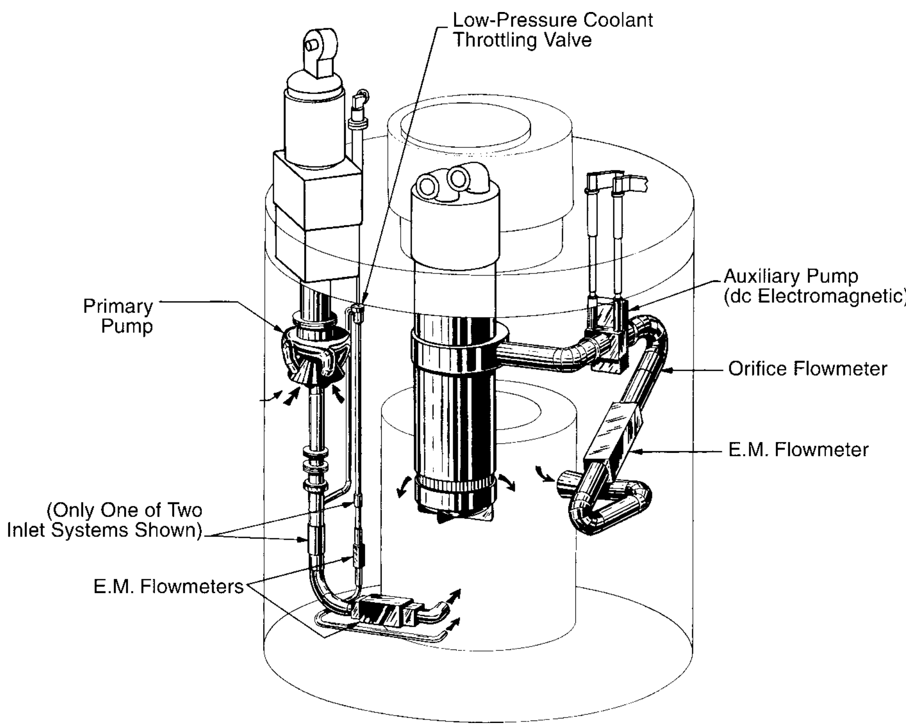


FIGURE 2-11. EBR-II PRIMARY PIPING AND COMPONENT ARRANGEMENT.

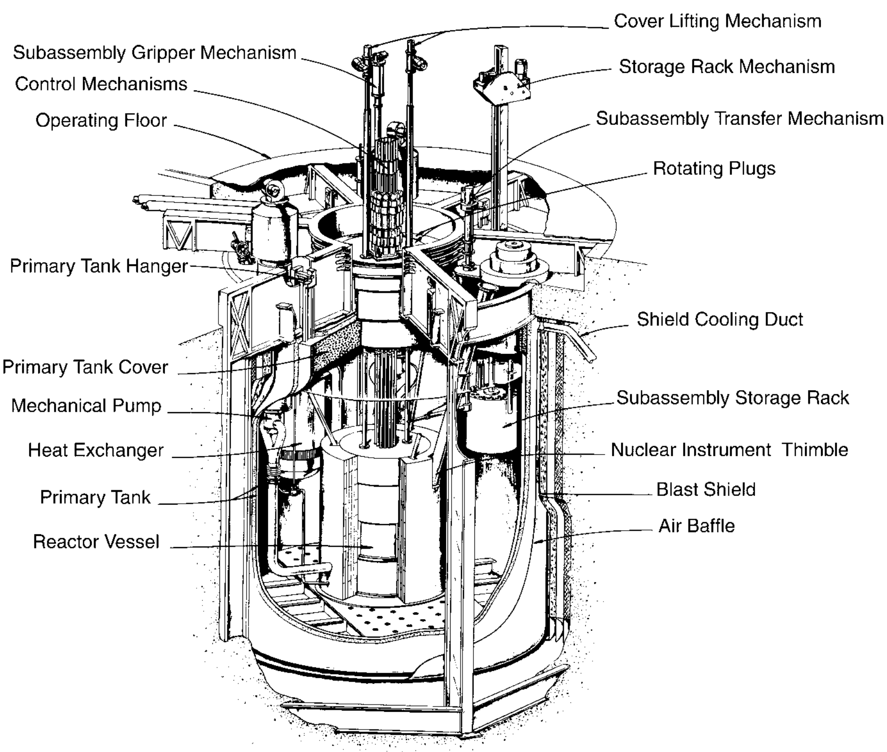


Figure 2-12. PRIMARY SYSTEM.

As a consequence, the absolute basic requirement for the EBR-II concept was to achieve fission product decay heat removal and shutdown cooling totally independent of heat removal by the secondary system. It required passive systems that would remove heat from the fuel and eventually transfer it to the atmosphere, bypassing the secondary system entirely. This was to be achieved without external power using natural convection of the liquid coolants involved and by natural convection of air.

Detailed analyses of a variety of reactor shutdown conditions identified situations which could jeopardize the initiation of natural thermal convection of sodium through the reactor. Under these conditions, the fuel could overheat before natural circulation of the sodium would begin. To avoid such situations, an auxiliary pump was installed in the outlet sodium line (as shown in Figure 2-11) which ensured low flow through the reactor at all times.

It included a direct current power supply to the pump but, in the event of failure of all power supplies, backup battery power would operate the auxiliary pump for at least 30 minutes before the batteries were discharged. This system ensured a reliable transition to natural convection circulation of sodium through the reactor no matter what sodium flow conditions existed at shutdown. This arrangement of the primary system ensured that under the most demanding circumstances the heat generated in the fuel by fission product decay would be removed and transferred to the bulk volume of sodium in the primary tank.

Under normal shutdown conditions, decay heat was transferred to the atmosphere through the secondary sodium system and the steam/feed water system. This transfer happened under controlled conditions that maintained the bulk sodium in the primary tank at the desired temperature.

Under abnormal conditions such as after shutdown where heat was not removed from the primary system through the normal power cycle, the heat was retained in the primary sodium. The 86,000 gallons of sodium provided a huge heat sink but the temperature would slowly rise if heat was not removed.

Two shutdown coolers were provided in the EBR-II primary system to remove this heat. Since this heat removal had to be provided reliably under all conditions, it was a passive system. The system operated by natural thermal convection, removing heat from the primary sodium and transferring it out of the reactor building to the atmosphere. The heat transfer medium was sodium-potassium eutectic alloy that was liquid at room temperature. (This alloy was the primary and secondary coolant for EBR-I for that reason.)

The sodium-potassium eutectic alloy flowed by natural thermal convection through a heat exchanger in the primary tank to an air-cooled heat exchanger in an air stack outside the reactor containment building. Heat was removed by natural convection of air.

To ensure reliable operation of this system, the shutdown coolers operated all the time. Continuous low heat removal was maintained by dampers in the air stack that restricted natural circulation of air through the stack. The damper was held closed by an electrically energized magnet. Upon receipt of a signal, or in the case of a power failure (a fail safe provision), the damper opened, the air flow through the stack increased by natural convection, and the heat removal rate increased (Figure 2-13).

The sodium-potassium eutectic alloy coolant circuit contained no valves and could not be shut off. The heat removal capability was reduced but not stopped by restricting heat removal from the sodium-potassium eutectic alloy to the air heat exchanger.

Figure 2-14 shows the temperature conditions that would result after reactor shutdown without heat removal from the primary sodium through the intermediate heat exchanger, both with one shutdown cooler and two shutdown coolers operating. Fission product decay heat removal would continue indefinitely and was not dependent upon a power supply of any kind.

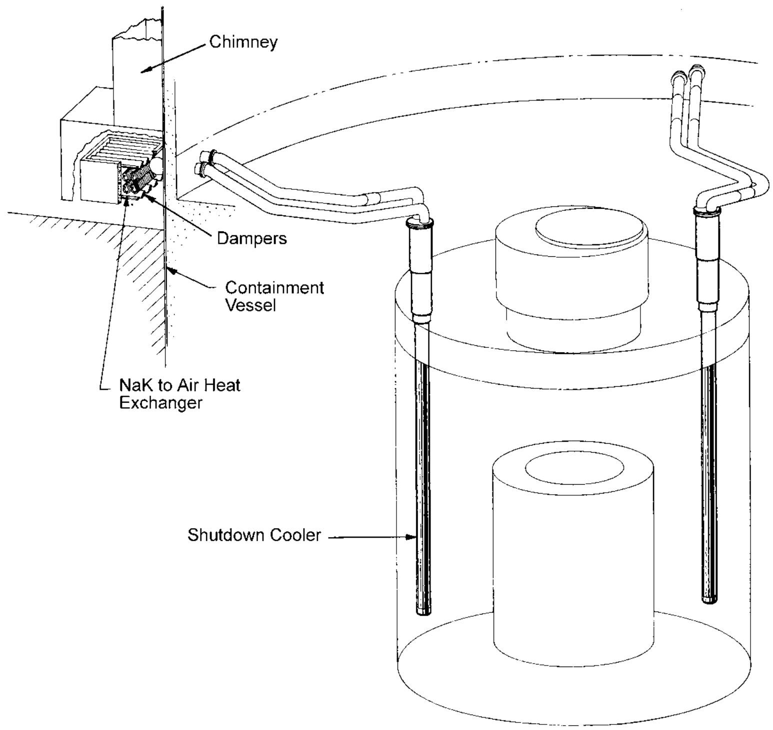
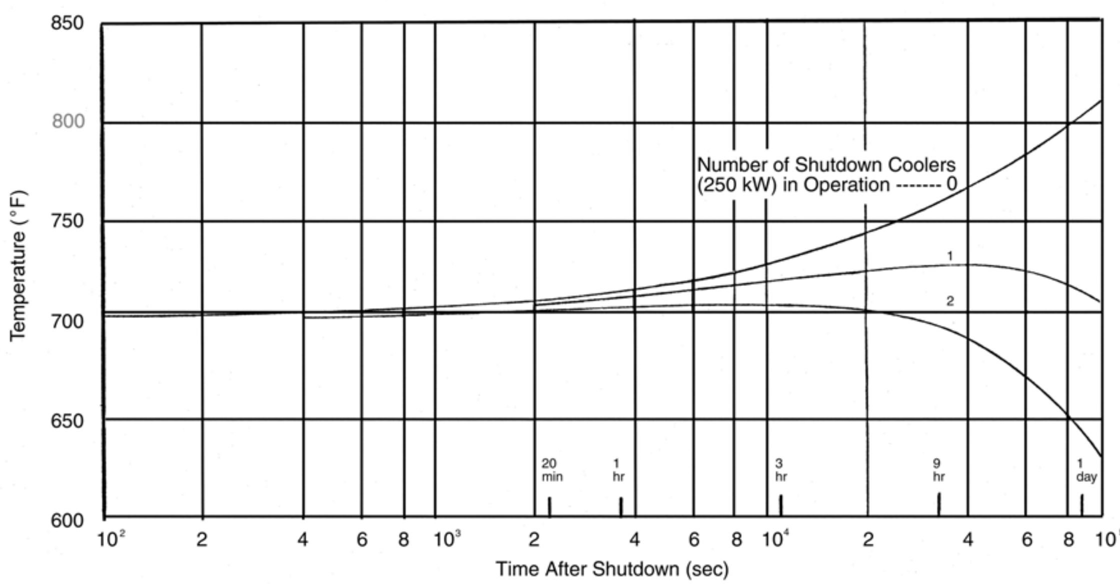


FIGURE 2-13. EBR-II SHUTDOWN COOLING SYSTEM.



Notes:

1. Total Heat Input is by Fission Product Decay Based on Initial Operation at Full Power
2. No Heat Loss to Secondary System (Through Heat Exchanger)
3. Total Parasitic Heat Loss = ~130 kW

FIGURE 2-14. PRIMARY TANK BULK SODIUM TEMPERATURE VS. TIME AFTER SHUTDOWN.

The very simple and basic EBR-II cooling concept was also very versatile and satisfied a variety of normal as well as off-normal conditions. It was a simple two-step process:

- Remove heat from the fuel and transfer it to the primary sodium
- Remove heat from the primary sodium and transfer it to the atmosphere.

This process not only applied to the sequences described above, but also at all times when the reactor cover was raised, and the reactor outlet piping and intermediate heat exchanger flow system were bypassed. This condition existed during fuel handling and other times when the reactor was open. The basic requirement for this simple process was to keep the fuel, which was the heat source, submerged in the primary sodium, the heat sink.

HEAT REMOVAL, TRANSFER, AND UTILIZATION FOR POWER GENERATION

Although in many respects the EBR-II concept reflected the requirements imposed by reactor shutdown considerations, it also demonstrated the technical feasibility of utilizing a sodium cooled fast reactor as an energy source for generating electricity. Power cycle conditions and requirements were applied to the power system components, while simultaneously ensuring that they would meet the shutdown requirements. Emphasis was placed on reliability of operation and serviceability of components.

A few parameters were set on the basis of judgment and broad objectives. For example, there was a desire to operate with super-heated steam. Steam conditions of 850°F and 1,250 pounds per square inch were selected because they were typical for small plants at that time and the capital cost of associated equipment was favorable. EBR-II was based on the goal that fuel costs for liquid metal cooled fast breeder reactor power plants should be low, and therefore thermal efficiency was not a primary consideration. Capital cost and reliable efficient operation would be more important in evaluating fast reactor power systems.

These considerations led to a 900°F primary sodium outlet temperature. A temperature rise through the reactor of 200°F appeared achievable since the reactor core was only 14 inches high. At a thermal power level of 62.5 megawatt thermal to achieve 20 megawatt electric the other variables such as primary sodium flow rate, secondary flow rate, steam flow rate fell into place.

Although the operating parameters were conventional, many of the components comprising the power system were unique and imposed special requirements.

The intermediate heat exchanger was designed to permit complete removal of the tube bundle. The systems were arranged so that the pressure of the secondary sodium in the tubes of the heat exchanger was higher than the pressure of the primary sodium outside the tubes. This ensured that in the event of a tube leak, non-radioactive secondary sodium would leak into the primary sodium, and not vice versa.

Of necessity, the inlet and outlet secondary sodium lines had to enter the top of the heat exchanger. The inlet cold secondary sodium was directed through a central pipe to a plenum and tube sheet at the bottom of the intermediate heat exchanger. The secondary flow was up through the tubes and the primary flow was down outside the tubes in a conventional counter flow design. In a vertical unit this provided that the heated fluid was flowing up and the cooled fluid was flowing down, the correct arrangement for sustaining natural thermal convection circulation.

The intermediate heat exchanger was not removed during the operating lifetime of EBR-II, but during construction it was verified that it could be removed. The intermediate heat exchanger and the permanent primary sodium piping were installed relatively early in the construction sequence of primary system component installation. The installation of the intermediate heat exchanger tube bundle is shown in Figure 2-15.

The intermediate heat exchanger tube bundle was then removed and stored. The large open nozzle for the intermediate heat exchanger was used as the personnel access to the primary tank during the installation of the balance of the primary system components (Figure 2-16). The final operation to close the primary tank involved the permanent installation of the intermediate heat exchanger.

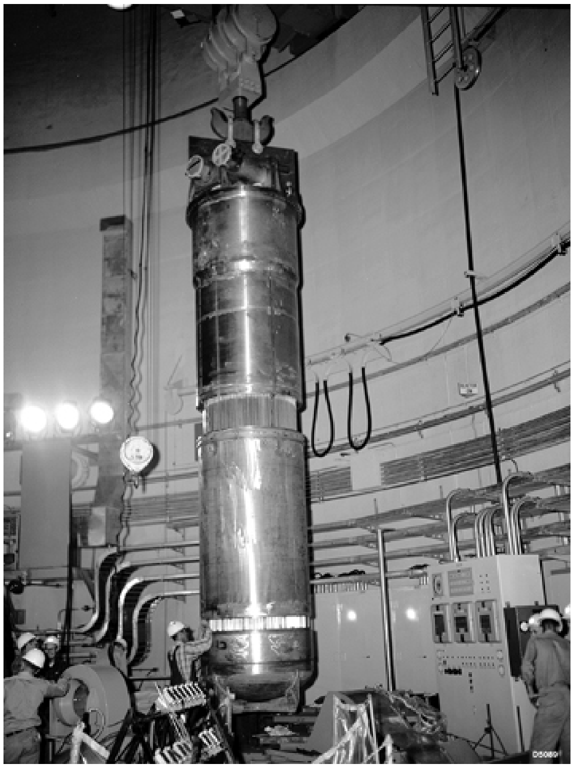


FIGURE 2-15. INSTALLED IHX TUBE BUNDLE (PRIOR TO INSTALLATION).

EVOLUTION AND IMPLEMENTATION OF THE SUBMERGED PRIMARY SYSTEM CONCEPT

The EBR-II submerged primary system concept evolved rather slowly; it was not discovered or invented in a spectacular stroke of genius. The process of identifying and evaluating operating characteristics of liquid metal cooled fast breeder reactors produced a variety of potential concepts.

The needs to achieve high power density for power operation and to accommodate the consequent high fission product decay heat were critical. The considerations involved in the use of sodium on a large scale as a heat transfer fluid were also a major factor in developing the concept. There was very little applicable experience available and the process involved the evaluation of ideas without the benefit of background experience or knowledge. Even those concepts based on more conventional systems required application of undeveloped technology.

Superimposed over all of these considerations was the recognition that this revolutionary reactor concept would require successful demonstration to achieve acceptance. Reliable, predictable operation was a mandatory objective of the project. All the options were evaluated on this basis and, even though a radical concept evolved, the process was conducted very conservatively.

To enhance the achievement of reliable plant operation, reliability, and serviceability of major components were extremely important. Major components and systems were placed into two basic categories: removable and non-removable. The non-removable components were expected to have a lifetime equivalent to the operating life of the plant, or the plant had to be able to operate without them.

As the reactor and primary system concept developed, a non-removable, permanent system evolved consisting of the primary tank, the reactor structure, and the primary sodium piping from the pumps to the reactor and from the reactor to the outer shell of the intermediate heat exchanger. The permanent system did not include the pumps or the intermediate heat exchanger tube bundle; they were removable. The reactor structure consisted of the lower grid-plenum, the cylindrical shell, the lower and radial neutron shield, and the reactor cover. The reactor cover fell into a somewhat different category because it was moveable, but not readily removable. All of the non-removable components were permanently attached to the inside of the primary tank.

The primary tank contained all of the components comprising the reactor and primary sodium system—non-removable as well as removable. Not only did the primary tank contain all of the primary system components, but it also contained the 86,000 gallons of primary sodium. The primary tank was double walled, with inert gas in the annulus between the two tanks. The outer tank was insulated. There were no penetrations or openings in the vertical cylindrical section or the bottom of either tank. All openings and penetrations into the tank were through the top cover. The bulk sodium level in the tank was maintained more than a foot below the underside of the top cover, and therefore there were no penetrations below the sodium level.

The primary tank was hung from the top structure by six hangers equally spaced to permit radial expansion of the tank. The early design of the hangers consisted of a double hinge arrangement shown in Figure 2-17.

FIGURE 2-16. OPEN NOZZLE FOR THE IHX (LOOKING UP).

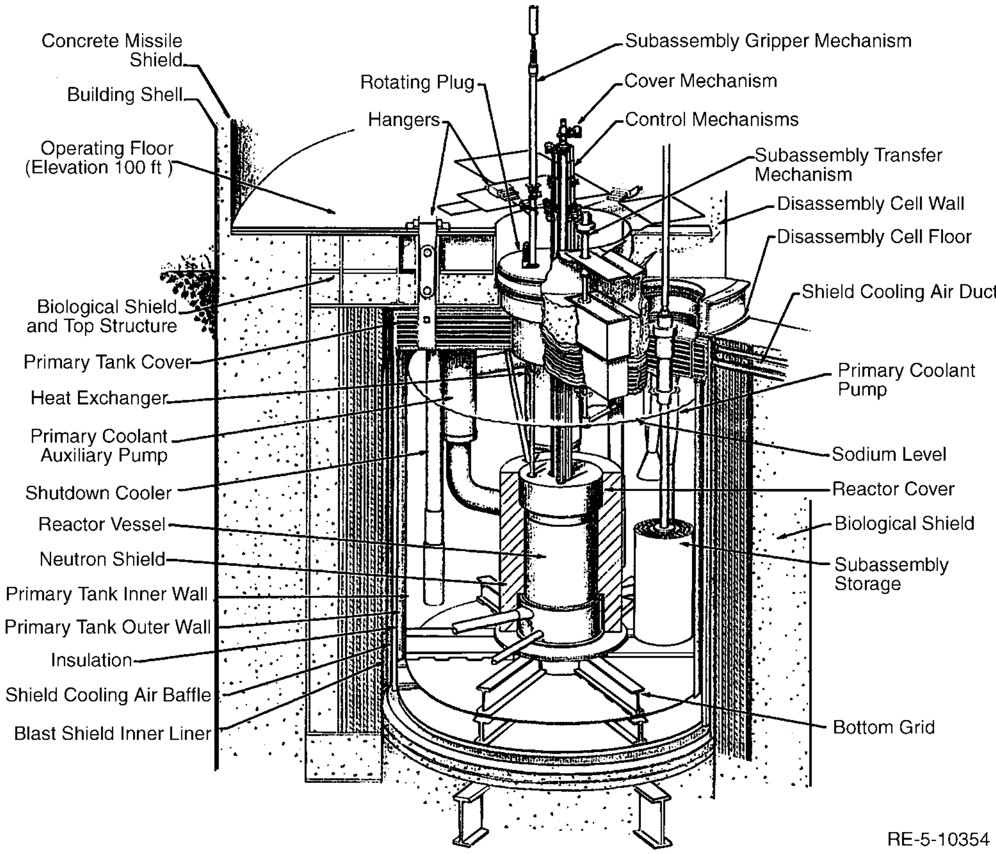


FIGURE 2-17. EBR-II PRIMARY SYSTEM (EARLY DESIGN).

This design was superceded by a roller hanger arrangement shown in Figure 2-12. The roller design permited inspection of the moving parts and measurement of movement during change in temperature of the primary tank. The rollers and support plates were actually removable and replaceable. The inspectability and replaceability of this design represented a significant improvement in reliability, even though no need for repair or replacement arose during the 40 years operating lifetime.

The primary tank and other major components were supported by a symmetrical structure shown in Figure 2-18. It was designed to not only support the total weight suspended from it, but also to survive a high energy release in the reactor and primary system.

The non-removable components, except the intermediate heat exchanger shell, were supported on the bottom of the primary tank. These loads were carried by the cylindrical section of the inner tank wall of the primary tank and the bottom of the inner tank. The details of this lower support structure and a cross-section of the reactor grid/plenum structure are shown in Figure 2-19.

Since the primary tank was hung from the top, it expanded vertically and radially as the bulk sodium temperature increased, and vice versa. This expansion occurred slowly because of the thermal capacity of the bulk sodium that established the tank temperature.

Because of thermal expansion considerations, the most position-sensitive components, such as the reactor and related control and fuel handling components, were positioned at and around the center of the primary tank. Less position-sensitive components were located out from the center, but provisions were made to accommodate movement resulting from thermal expansion.

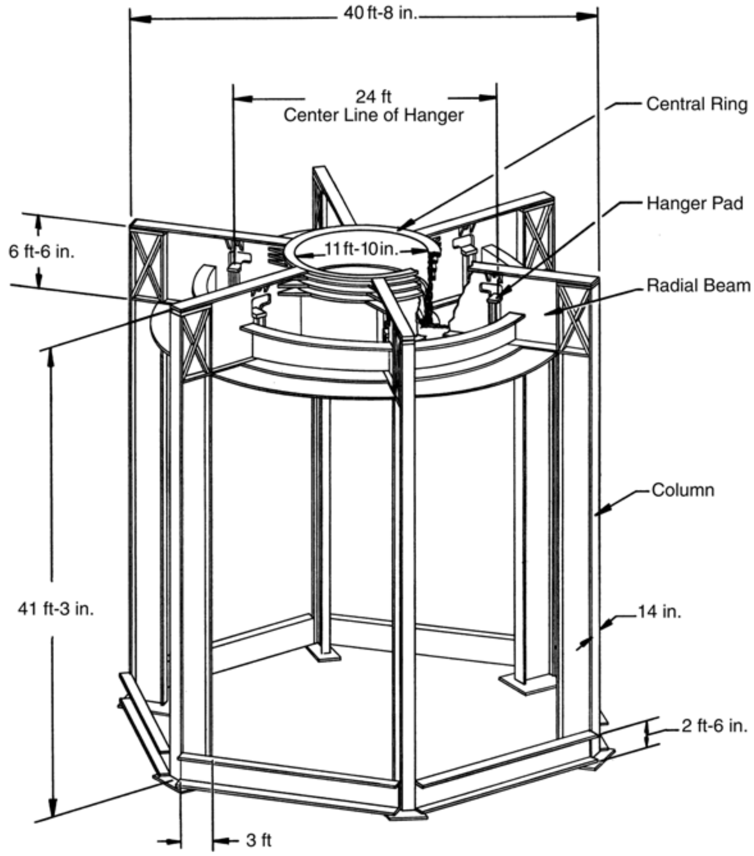


FIGURE 2-18. PRIMARY TANK SUPPORT STRUCTURE.

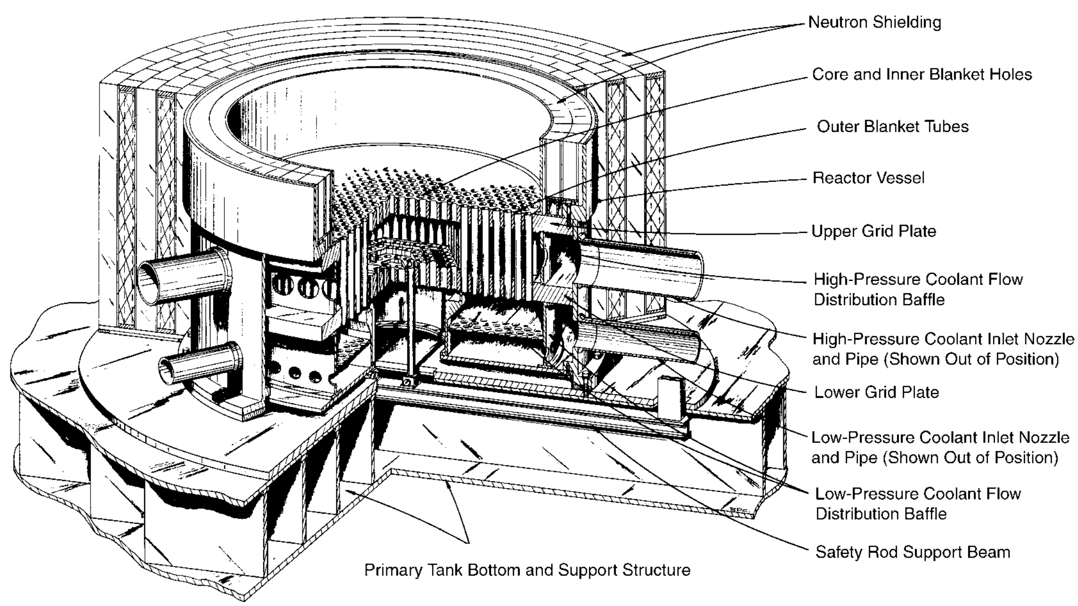


FIGURE 2-19. PICTORIAL OF REACTOR VESSEL GRID ASSEMBLY.

As can be seen in Figure 2-12, the top cover of the primary tank incorporated a large number and variety of penetrations. Most of these housed cylindrical components such as instrument thimbles and were readily removable. Of special interest were the pumps and intermediate heat exchanger, which involved extensive operations for removal. To accommodate pump removal, which in a conventional system involved cutting pipes, a mechanical ball and seat detachable connection was developed as shown in Figure 2-20. Although this was not a leak-tight joint, the leakage was permissible since the sodium leaked into the bulk sodium, which was the intake supply to the pump.

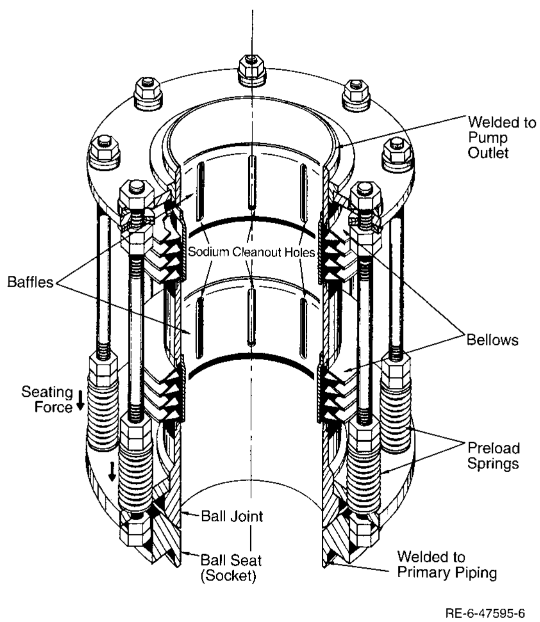


FIGURE 2-20. BALL-JOINT CONNECTOR.

An interesting aspect of the EBR-II concept development involved the primary sodium pumps. At the time, there was very little experience with mechanical pumps and much concern about their reliability. EBR-I employed direct current electromagnetic pumps and early development of alternating current electromagnetic pumps was very promising; the U.S. Naval Reactor Program supported much of this development. The original EBR-II concept incorporated direct current electromagnetic primary sodium pumps. Tests were conducted to verify that operation submerged in sodium was feasible. Their major drawback was the requirement for very high current at extremely low voltage.

During this period, an advance was made in mechanical pumps with the development of the hydrodynamic bearing, which was being used to pump fluids with poor lubricating qualities. This type of bearing proved to be successful with sodium, and mechanical pumps were selected for use in the EBR-II primary sodium system. However, as a precaution, the rectangular shaped penetrations in the top cover were retained so that the mechanical pumps could be replaced with direct current electromagnetic pumps if necessary.

The submerged primary system concept was a radical departure from conventional piped system arrangements and was even more revolutionary because the fluid in which the system was submerged was high-temperature sodium. There was much concern about the feasibility of the concept. The concept evolved because it provided a positive and effective response to two basic requirements:

- Absolute reliability of reactor cooling, particularly for all possible scenarios for fission product decay heat removal
- A realistic process for refueling the reactor in spite of the requirement that these operations be performed in a very difficult and hostile environment.

The absence of extensive coolant piping, the compactness of the system with resultant minimal radiation sources, and system compatibility with a unique fuel cycle were other benefits that contributed to reliability.

The concept became more acceptable, and then preferable as the first two requirements were met and then other benefits were identified. One benefit frequently overlooked was that sodium leaks could be accommodated because they returned the sodium to the system.

Because there was no applicable experience to draw upon as the EBR-II concept was developed, the process really became one of addressing a series of "what ifs." As scenarios evolved, program teams evaluated virtually every conceivable application of Murphy's Law. Interestingly, this iterative process served to strengthen the conviction that the system concept could work.

The next chapter focuses on the application of the exhaustive concept planning to the EBR-II systems and components.

CHAPTER 3 — DESCRIPTION OF EBR-II SYSTEMS AND COMPONENTS

This chapter describes the EBR-II. It consisted of four major, integrated, functional systems:

1. **THE PRIMARY SYSTEM** — the reactor and associated equipment, and the primary sodium cooling system. The energy source and the heat removal system.

2. **THE SECONDARY SYSTEM** — the intermediate sodium heat transfer system. The non-radioactive heat delivery system, and the heat source for steam generation.

3. **THE STEAM ELECTRIC SYSTEM** — a conventional superheated, condensing turbine-generator system, which provided the end product — electricity.

4. **The Fuel Recycle System** — the system for decontaminating and manufacturing the nuclear fuel.

The first three systems comprised the power system. The heat produced in the reactor was removed by the primary sodium system and transferred to the secondary sodium system in the intermediate heat exchanger. From the secondary system, the heat was transferred to the steam system in the steam generator to produce superheated steam, which was then delivered to a conventional condensing turbine at 850°F and 1,250 pounds per square inch gauge. A simplified flow diagram of the power system is shown in Figure 3-1. A temperature-enthalpy diagram is included as Figure 3-2.

These systems were housed in four plants and supporting facilities and structures as shown in Figure 3-3. The plants were designated as follows:

The Reactor Plant contained the reactor and primary sodium cooling system and supporting services to these facilities. It consisted of a containment building designed to contain any accidental release of radioactive material within the building. It was interconnected to the Fuel Cycle Facility and the Power Plant.

The Sodium-Boiler Plant contained the entire secondary sodium system, including the steam generator, except for the piping to the Reactor Plant and the intermediate heat exchanger, which was installed in the primary tank. The building had two wings — the sodium wing and the boiler wing. It contained unique features reflecting the incompatibility of sodium and water/steam.

The Sodium-Boiler Plant was somewhat isolated within the system complex. It was linked to the Reactor Plant by 75 feet of sodium lines and to the Power Plant by 200 feet of steam and condensate lines. The building contained only the minimum facilities for operation and was not normally occupied by operating personnel.

The Power Plant contained the turbine generator and associated equipment and the control room for the reactor and power cycle. It was interconnected to the Reactor Plant by means of one air lock to permit personnel access to the Reactor Plant. The building was of conventional construction.

The Fuel Cycle Facility contained two shielded cells for disassembly, processing, and manufacture of fuel elements and subassemblies, and supporting facilities for these operations. It also contained the inert-gas storage facilities, the sodium equipment cleanup cell, and exhaust ventilation system and the stack for the exhaust from the Fuel Cycle Facility and Reactor Plant. It was interconnected to the Reactor Plant.

An additional building, the Laboratory and Service Building, located adjacent to the Fuel Cycle Facility provided supporting analytical facilities for control of the fuel cycle and process operations. It also provided facilities for personnel and supporting services.

PRIMARY SYSTEM

The Primary System (Figure 2-12) was housed in the Reactor Plant and included the following:

- Reactor
- Subassemblies
- Reactor Vessel Assembly
- Primary Cooling System
- Shutdown Cooling System
- Neutron Shield
- Counters, Chambers, and Instrument Thimbles

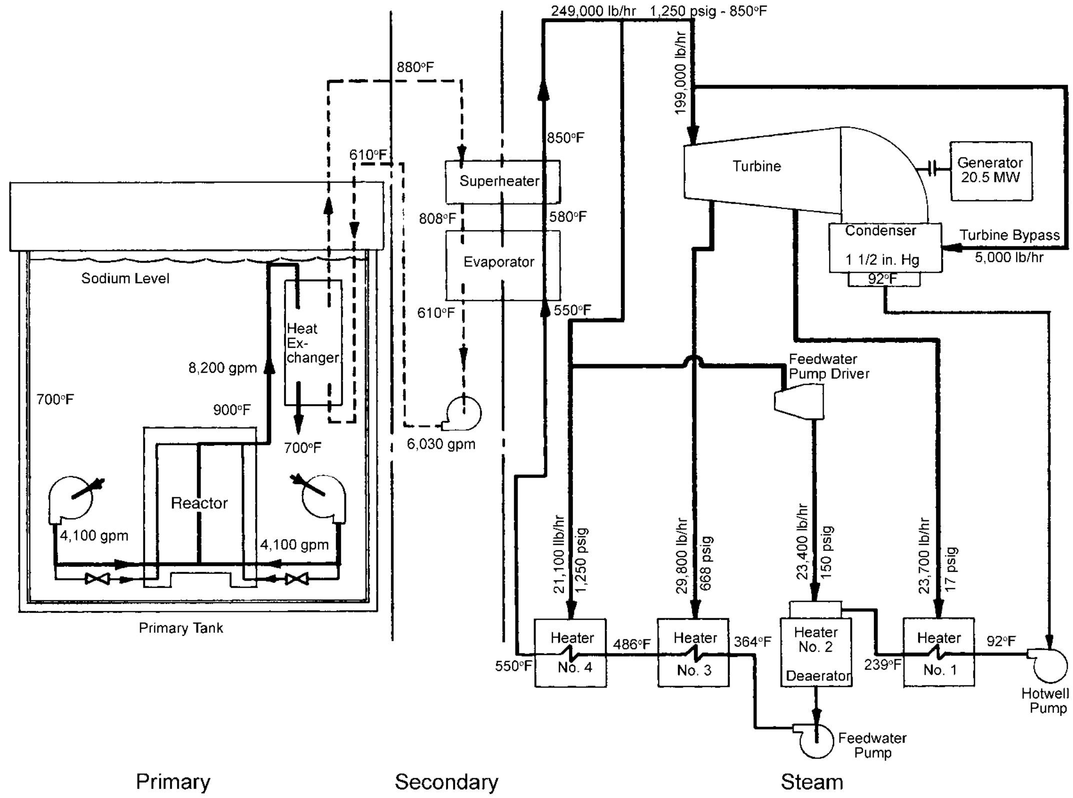


FIGURE 3-1. EBR-II SKELETON FLOW DIAGRAM.

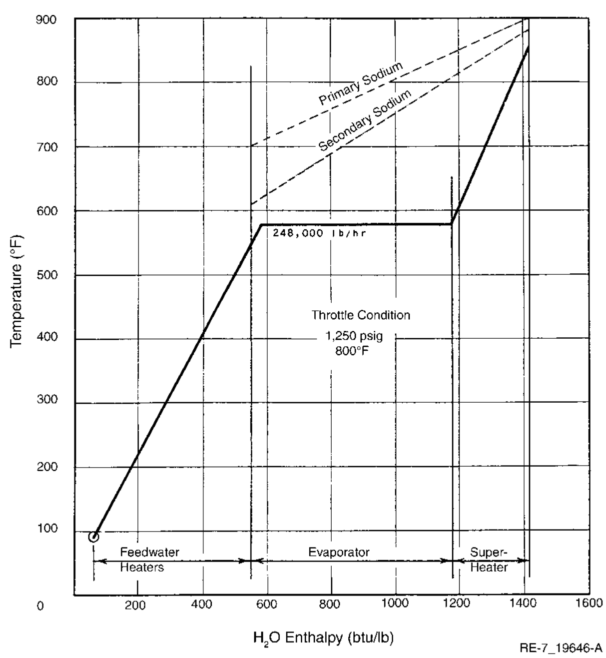


FIGURE 3-2. TEMPERATURE-ENTHALPY DIAGRAM.

- Control and Safety Drive Systems
- Fuel Handling System
- Primary Tank and Biological Shield
- Primary Sodium Purification System
- Inert Gas System.

The reactor, the primary sodium pumps and piping, the heat exchanger, and the fuel handling system were contained in the primary tank submerged in sodium, as shown in Figure 2-12. Coolant was pumped directly from the bulk sodium in the primary tank to the reactor, and after flowing through the reactor, passed through the heat exchanger and back to the bulk sodium. This very simple flow system is shown in Figure 2-11. This submerged concept was employed for the following reasons:

1. The arrangement contributed significantly to the reliability of the primary coolant system. A high degree of integrity could be constructed into the primary tank, since it was of relatively simple design. As an added safety measure, it

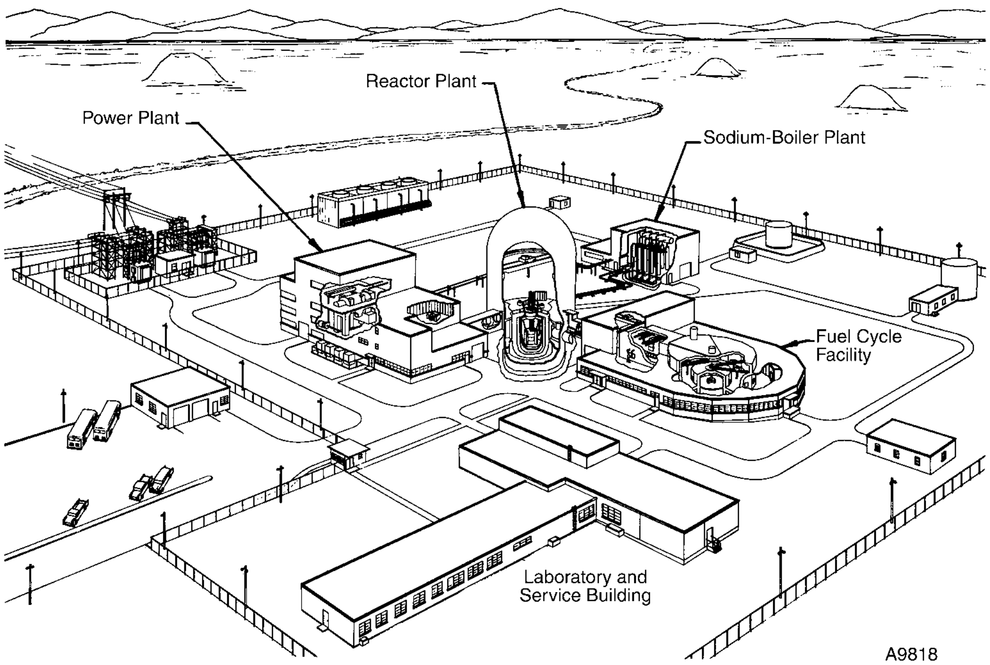


FIGURE 3-3. EBR-II PLANT ARRANGEMENT.

consisted of double-wall construction (a guard tank surrounding the primary tank). Because the entire coolant system was flooded with sodium to a level approximately 10 feet above the top of the reactor loss of coolant for the reactor was virtually impossible. Even if both primary tank walls were to fail, the free volume between the guard tank and the liner of the biological shield was sufficiently small to maintain the sodium level above the top of the reactor.

2. Since the reactor was intended to demonstrate operation suitable for a central station Power Plant, the replacement of fuel was accomplished in a short time. Shortly after reactor shutdown, the heat generation in the fuel was high, and reliable cooling was provided. This was accomplished by handling the fuel subassembly submerged in sodium. The fuel was cooled by natural convection of sodium through the subassembly, and fuel handling could begin immediately after reactor shutdown. The fuel subassemblies were moved to a fuel storage rack within the primary tank where they continued to cool, by natural convection of the sodium, until removed for processing.

3. Leak tightness of the primary coolant system piping was not required. Small amounts of leakage were permissible, since the leakage was internal. A small amount of leakage occurred at the connections between the pumps and the reactor, between the reactor tank and the reactor cover, and around subassembly nozzles.

4. The heat capacity of the very large mass of bulk sodium, approximately 620,000 pounds provided considerable thermal inertia to the primary system. It prevented rapid temperature transients in the primary sodium coolant reactor inlet temperature, and it added reliability to the shutdown cooling system.

5. Maximum integrity was provided with regard to containment of radioactive sodium. The entire radioactive coolant system, with the exception of the single, small, sodium cleanup circulation circuit, was confined within the primary tank.

6. Essentially all of the radioactivity in the Reactor Plant was confined to the primary tank and, therefore, only the primary tank, and the single circuit referred to in No. 5 above, required shielding. Shielded equipment cells and pipe galleries were eliminated.

7. Auxiliary heating of the primary system sodium to prevent freezing was simplified

since the entire system was heated as a unit. The individual components and pipes were in an environment of sodium, and the entire system was at one temperature.

REACTOR

The reactor was divided into three main zones: core, inner blanket, and outer blanket (Figure 2-2). Twelve control rods were located at the outer edge of the core, and two safety rods were located within the core, as shown. Each zone comprised a number of hexagonal subassemblies. The three zones were established by the lower grid-plenum structure, which used three different diameter holes for accepting the three types of subassemblies (including control and safety rods), which comprise the zones as shown. The number of subassemblies comprising the three zones as shown in Figure 2-2 are tabulated below in Table 3-1.

The basic minimum core volume, including control rods and safety rods, consisted of a total of 61 subassembly units (rows 1–5). This represented the minimum core volume for which reactor performance was evaluated and the minimum configuration that was used in the reactor. To provide flexibility of operation and to accommodate variations in core loading, which was practiced throughout the EBR-II operating lifetime, fuel subassemblies identical to those in the core zone were provided and could be installed in the first row of the inner blanket.

TABLE 3-1. Subassembly distribution in reactor.

Core	47[a] (47 to 59)[b]
Safety	2
Control	12
Inner Blanket	66[a] (66 to 54)[b]
Outer Blanket	510
Total	637

a. Minimum volume core.

b. Normal permissible core volume range (to accommodate experimental program).

Analyses were performed for cores incorporating from 1 to 12 of these special inner blanket fuel assemblies. A basic configuration of six of these special subassemblies was considered the nominal reactor loading. The possible arrangements considered where the additional inner blanket type fuel subassemblies were loaded and the sequence and location, from number 1 to 12, were specified. The comparable loading patterns are shown in Figure 2-2. The range of subassembly units in each zone is also shown in Table 3-1. This arrangement provided great flexibility and was extremely useful over the operating lifetime of the reactor.

The core, including the control and safety rods, had an equivalent radius of 9.92 inches (24.17 centimeters) and a height of 14.22 inches (36.12 centimeters); a total core volume of 66.3 liters. The core volume was varied frequently over the operating lifetime of the plant; it was easily increased by placing the special fuel subassemblies in the first row of the inner blanket as described above. The coolant flow control system easily accommodated this arrangement by providing appropriate coolant entry holes in the lower adapter of the subassemblies. Also, using elements with longer fuel sections increased the core height.

The 12 control rods and the 2 safety rods consisted of modified movable fuel subassemblies and were a part of the core zone. The rods, plus their stationary thimbles, comprised the control and safety subassemblies. The external dimensions of the thimbles were identical to the core and blanket subassemblies and the lattice spacing for all units was identical. The reactor could be controlled by moving the control rods in their thimbles in a vertical direction, thus moving fuel into or out of the core.

The safety rods were not a part of the normal reactor operational control system but were maintained in their "full-up" position, or maximum reactive position, at all times during reactor operation. This position was also maintained during fuel handling operation when the control rods were disconnected from their drives and were in their least reactive position.

SUBASSEMBLIES

A single subassembly size was employed throughout the reactor, resulting in a close-packed reactor geometry. The hexagonal subassembly tube was 2.290 inches across external flats with a 0.040-inch wall thickness. The subassemblies were spaced on a triangular pitch of 2.320-inch center distance. The nominal clearance of 0.030 inches between each subassembly permitted removal of the units from the reactor.

Each subassembly was located and supported at the bottom by the combination support grid and inlet coolant plenum (Figure 3-4). The heat generated in the fuel, or blanket material was removed by sodium flowing up through the subassemblies and around the fuel and blanket elements. To accommodate the very large range of flow rates required, two parallel flow systems were employed. A high-pressure system supplied the core and the inner blanket, while a low-pressure system supplied the outer blanket. The two systems had separate inlet plenum chambers as shown in Figure 2-19.

The upper end of each subassembly was identical, and the same handling and transfer devices accommodated all subassemblies. The lower adapters were of different size to differentiate the three types of subassemblies, and were of different configuration to accommodate the two coolant systems (Figure 3-5). Each subassembly contained a number of fuel elements, and/or blanket elements, of size and shape appropriate to the particular type of subassembly.

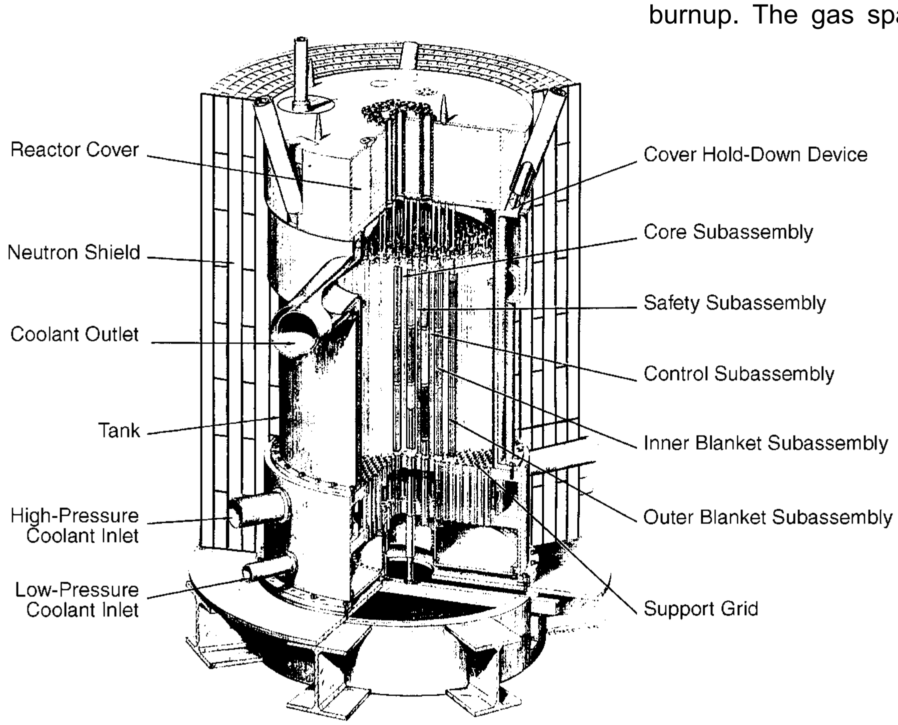


FIGURE 3-4. EBR-II REACTOR.

The core subassembly (Figure 2-1) comprised three active sections: upper blanket, core, and lower blanket. The core section consisted of 91 cylindrical fuel elements spaced on a triangular lattice by a single, helical wound wire on the outside of each element. The elements were supported within the subassembly and fastened at their lower ends to a support grid. The fuel elements (Figure 3-6) were pin type, consisting of a right circular cylinder of fuel alloy (0.144-inch diameter by 14.22 inches long) fitted into a thin-walled, stainless steel tube. The coolant flowed along the outside of the element tube.

The fuel pin was contained in a stainless steel tube (0.009-inch wall thickness by 0.174-inch outside diameter). The resultant annulus between the pin and the inside of the tube (0.006 inch) was filled with static sodium to provide a thermal bond. The sodium bond extended a nominal 0.6 inch above the top of the fuel pin. An inert gas space was provided above the sodium to accommodate expansion of the sodium. The fuel element tube was welded closed at each end. The fuel pin design evolved later to allow for higher fuel burnup. The gas space volume and the sodium bond annulus were increased to accommodate more fission gas and fuel swelling.

The individual fuel elements were contained within the hexagonal subassembly tube. They were fastened to the subassembly at their lower end by hooking to a parallel strip grid, as shown in Figure 2-1. The upper ends of the fuel elements were unrestrained to permit free axial expansion of the fuel element.

The upper and lower blanket sections were identical in construction and each consisted of 19 pin-type elements also spaced on a triangular lattice. The unalloyed depleted uranium pins were 0.3165 inches in diameter and totaled 18 inches long. They were similar in geometry to the fuel elements, being a

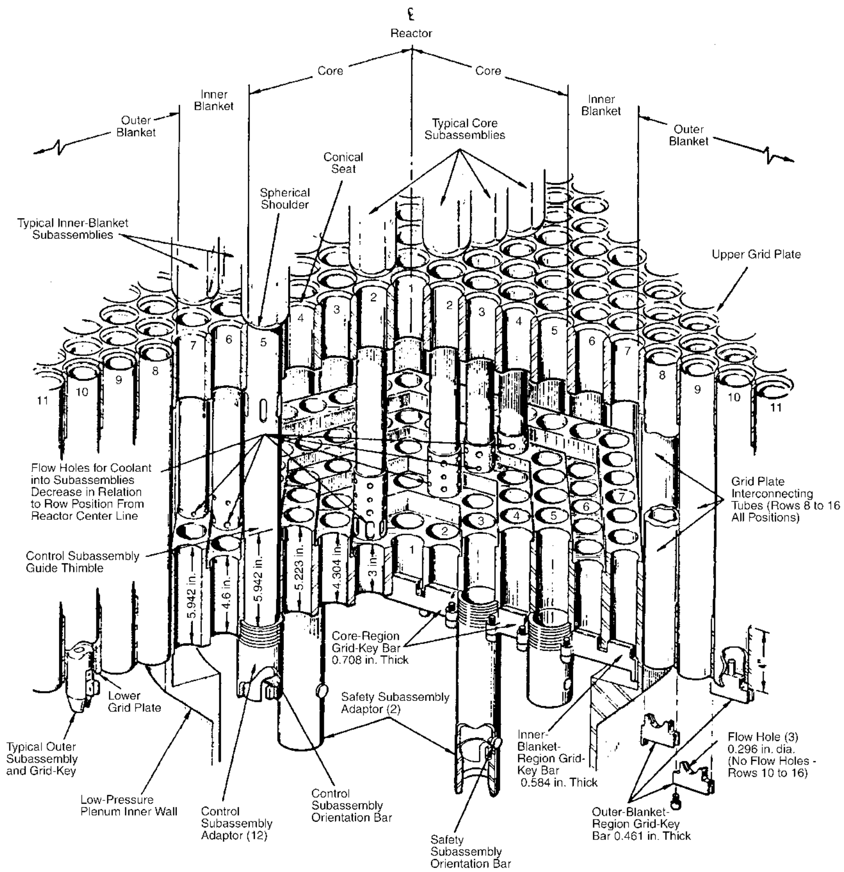


FIGURE 3-5. DETAILS OF GRID-PLENUM ASSEMBLY.

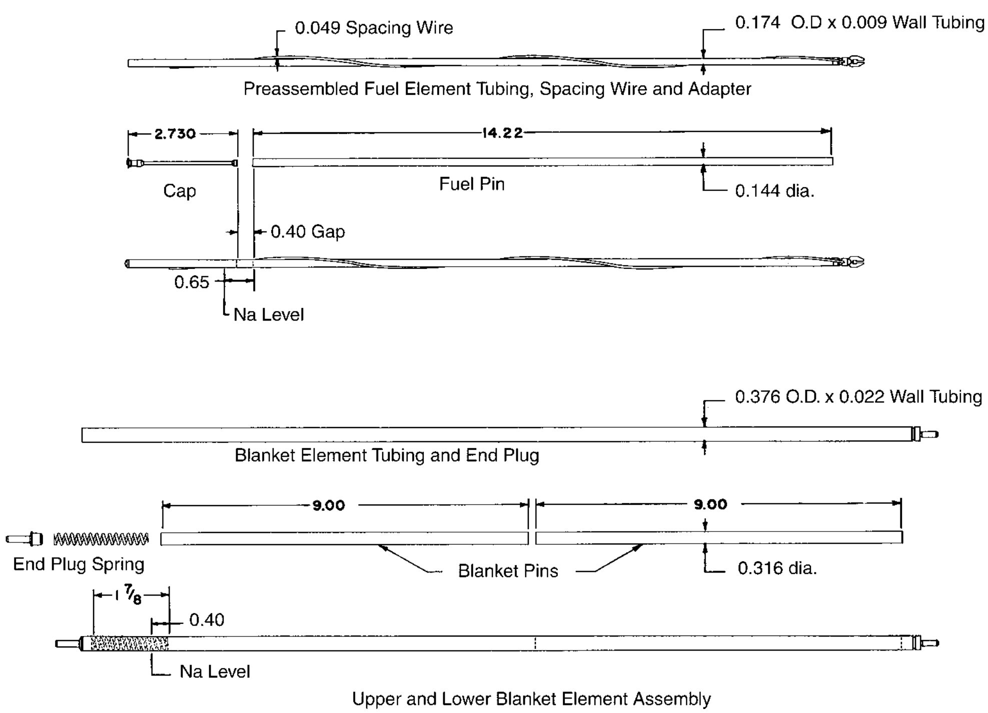


FIGURE 3-6. CORE SUBASSEMBLY ELEMENTS.

loose fit in the blanket element tube that consisted of a 0.370-inch outside diameter by 0.022-inch wall thickness. The 0.008-inch annulus was filled with sodium to provide the necessary thermal bond. Later core assembly designs abandoned the axial blanket sections. The details of the upper and lower blanket elements are shown in Figure 3-6.

The blanket elements were positioned in the subassembly by a grid structure at the lower and upper ends. They were fixed at the lower ends to the grid structure, while a grid also positioned the upper ends, permitting axial expansions but no other movement. Since the blanket elements were positioned at each end, no spacer provisions were made along the length of the blanket elements.

The lower adapter of the fuel subassembly engaged the reactor grid, and contained holes through which the coolant entered the subassembly from the high-pressure inlet coolant plenum chamber as shown in Figure 2-1.

The inner blanket subassembly (Figure 3-7) was made up of 19 cylindrical blanket elements spaced on a triangular pitch and contained in the hexagonal subassembly. The active blanket section consisted of depleted uranium cylinders (0.433-inch diameter) totaling 55 inches in length. They were contained in a stainless steel tube 0.493 inch in outside diameter with a 0.018-inch wall thickness. The resultant 0.012-inch annulus was filled with static sodium to provide a thermal bond. The sodium extended a nominal 2 inches above the top of the uranium, with a 4-inch argon gas expansion region above the sodium. The end closures were welded to provide a sealed unit. Flow distribution strips were included in the outer row of the elements to reduce the sodium flow in the peripheral flow channels to minimize over cooling.

The lower adapter of the inner blanket subassembly was similar to, but smaller in diameter than, the core subassembly. The inner blanket subassemblies also engaged the high-pressure inlet coolant plenum chamber in the reactor grid, as shown in Figure 3-7.

The outer blanket subassembly differed from the inner blanket subassembly in the design of the lower adapter and the design of the flow distribution strips. The lower adapter was arranged to engage the reactor grid in the low-pressure inlet plenum chamber. The two different lower adapters employed in the blanket assemblies are shown in Figure 3-7.

In addition to providing a low-pressure coolant supply to the outer blanket subassemblies, larger flow distribution strips were used in the peripheral flow channels to further reduce the coolant flow to match the lower power density in the outer blanket. The flow distribution strips in both the inner and outer blanket subassemblies are shown in Figure 3-8.

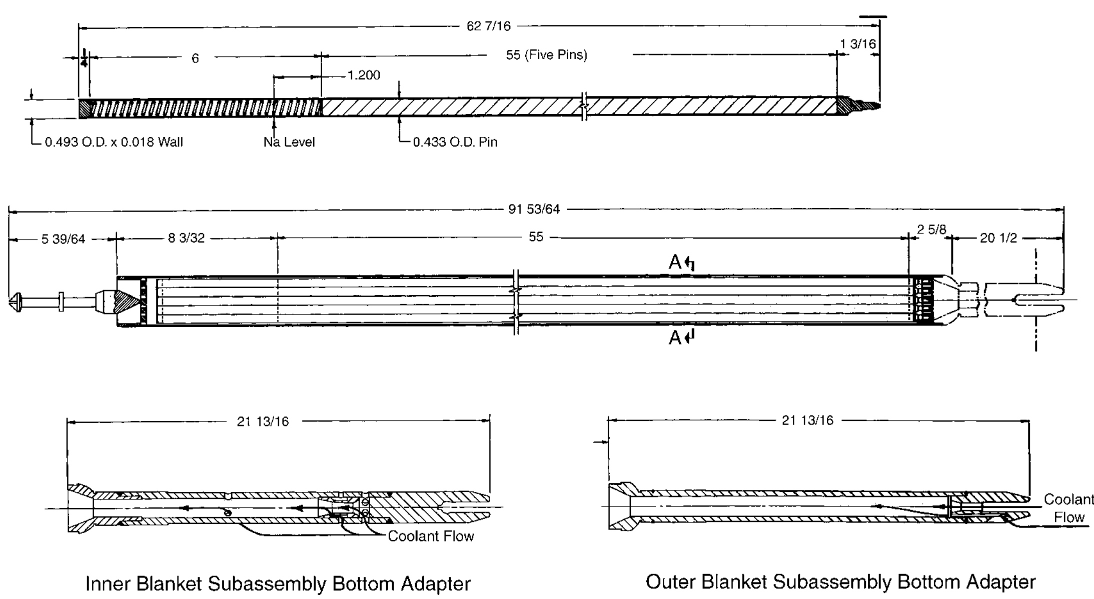


FIGURE 3-7. INNER BLANKET AND OUTER BLANKET SUBASSEMBLIES.

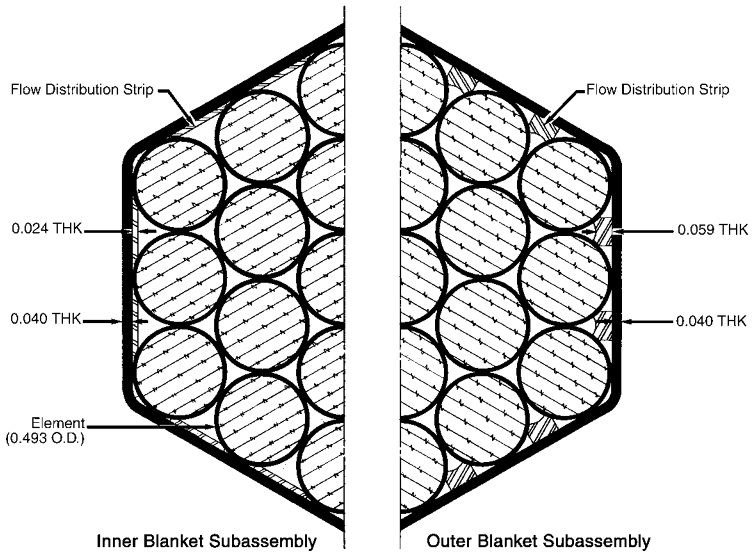


FIGURE 3-8. INNER BLANKET AND OUTER BLANKET SUBASSEMBLIES CROSS SECTIONS.

The control subassembly (Figure 2-6) consisted of a control rod and a guide thimble. The guide thimble occupied a unit lattice identical to those occupied by the various subassemblies.

Twelve identical control rods were employed to provide the operational control of the reactor. In later phases of the program, the number was reduced to as low as eight to permit up to four special test and/or irradiation assemblies to be installed in control rod positions. The control rod consisted of a modified fuel subassembly with a core section comprised of 61 fuel elements identical to the 91 fuel elements contained in the fuel subassembly. The control rod was encased in a hexagonal tube 1.908 inches across flats, which was smaller than the hexagonal guide thimble tube by the equivalent of one row of fuel elements. The control rod did not contain an axial blanket. A void section equivalent in height to the reactor core was provided above the fuel section of the control rod as shown in Figure 2-6.

During operation, this void section was filled with coolant sodium flowing through the control rod. A reflector section of solid steel, except for flow passages for the coolant, was located immediately above the void section. Reactor control was effected by vertical movement of the control rod, adjusting the proportion of fuel or void (sodium) in the core region of the reactor. As discussed earlier, the EBR-II reactor control concept was influenced by the desire to demonstrate high neutron efficiency to demonstrate the potential for maximizing breeding ratio. EBR-II did demonstrate the feasibility of operating the reactor by moving fuel and avoiding the use of parasitic absorber to achieve adequate control.

Subsequent operations of EBR-II demonstrated the use of a combination of absorber and fuel to increase the effectiveness of the control system. The absorber was located above the fuel section (the region of void described above). The upper end of the control rod was equipped with an adapter section identical to the subassemblies and was used for attachment to the control drive unit as well as the fuel gripper unit for unloading. The lower end of the control rod below the fuel section consisted of a cylindrical tube that also contained a steel reflector section. Bearings were provided on this lower section, which provided the guide between the control rod and the guide thimble.

The control rod was cooled in a similar manner to the core subassemblies by sodium delivered from the high-pressure sodium coolant system. Sodium entered through holes in the upper end of the lower adapter of the thimble, and through a second set of holes in the lower end of the control rod. The holes in the thimble section were above the lower bearing of the control rod throughout the control rod travel. The lower end of the thimble was open, and the lower control rod bearing served as a flow restriction to minimize sodium leakage from the bottom of the thimble. The primary system sodium pressure acted across the lower end of the control rod, and therefore exerted a downward force on the control rod. This downward force opposed the lifting force due to the pressure drop of the coolant flowing through the control rod, similar to the arrangement in the fuel subassemblies.

Since the vertical position of the control rods in the reactor varied relative to the stationary reactor core, the heat generation within the control rod was also variable. The coolant flow through the control rod was established to accommodate the maximum heat generation (i.e., with the control rod fully inserted in the reactor). If a constant coolant flow had been employed, the temperature rise in the coolant would have decreased as the control rod was lowered out of the reactor and the heat generated in the control rod decreased. This would have resulted in considerable degradation of the outlet sodium temperature from the control rod and in the mixed coolant temperature from the reactor.

To avoid this condition, an arrangement of the control rod and guide thimble coolant inlet holes provided variable orificing proportional to the position of the control rod in the reactor. This was accomplished by the relative size and locations of the coolant holes in the guide thimble and in the control rod. The coolant flow through the control rod varied with its vertical position in the reactor because the coolant flowed from the high-pressure plenum, through the holes in the thimble, at the top of the plenum, and then through the holes in the lower end of the control rod. The coolant flow path was shortest when the control rod was up and longest when the control rod was down.

This system did not provide precise control of coolant temperature, and the control rods were overcooled, but not to the extent that would have existed in a constant flow system. The flow reduction through the control rod was determined experimentally to be about 35 percent from control rod full up to full down.

A flow twister was installed in the void section immediately above the core section of each control subassembly to reduce the temperature differentials in the control rod hexagonal tube and, therefore, to minimize bowing of the control rod within its thimble. Upon leaving the core section, the hotter coolant flowed along the inside surface of the control rod facing the center of the reactor. The coolant was rotated approximately 180 degrees to the opposite surface by the flow twister. Thus, exposure of the opposite surface to the higher-temperature sodium tended to reverse any bowing of the rod. The flow twister did not introduce any significant pressure drop in the coolant flow. The coolant flow provisions for the control rods are shown in Figure 2-6.

The control rod was removed from the reactor by the fuel handling system in the same manner as the various subassemblies. The same considerations of irradiation damage and fuel recycling that applied to fuel subassemblies also applied to the control rods. The guide thimble was also removable from the reactor in the event of damage. It was locked in the lower reactor grid by a latch that was engaged by rotating the thimble. Rotation of the thimble was normally prevented by the six subassemblies that surround it. To remove or install a thimble it was necessary to first remove the six adjacent subassemblies and replace them with special modified scalloped hexagonal replacements that fill the subassembly space but permitted the thimble to be rotated. This special procedure was used infrequently.

The safety subassembly (Figure 3-9) consisted of a safety rod and a guide thimble. The safety rod and thimble were essentially identical to the control subassembly except for modifications at the lower end. Two safety rods were incorporated in the reactor and located as shown in Figure 2-2. The safety rods were not a part of the normal reactor operation control system. They were fully inserted in the reactor, in their most reactive position, at all times during reactor operation and fuel handling. The purpose of the safety rods was to provide available negative reactivity when the reactor was shut down and the control rods were disconnected from their drives. Their primary purpose was to provide a safety device during reactor fuel loading operations. The two safety rods were attached to the safety rod support beam located below the reactor structure and connected to two vertical drive shafts located outside the fuel transfer system and operable during refueling operation (Figure 2-19 and Figure 3-9).

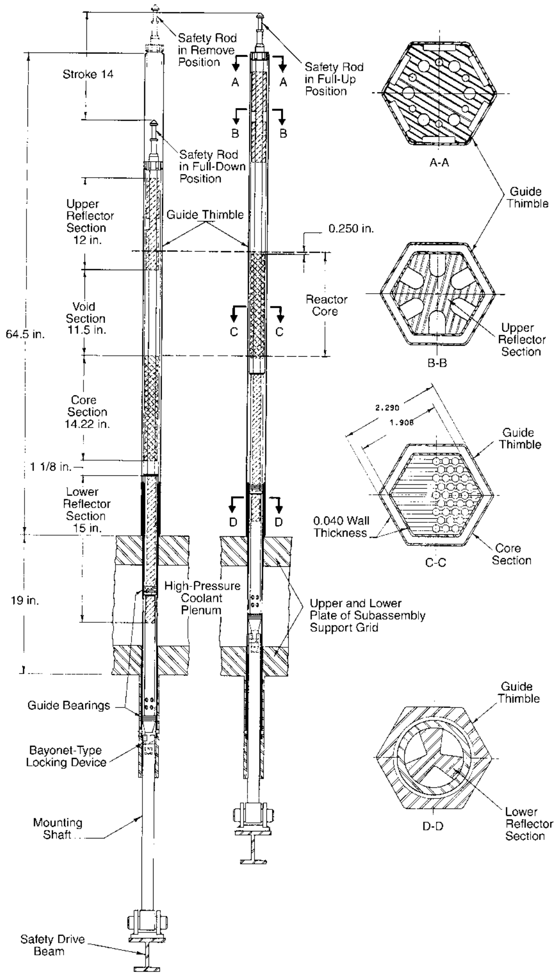


FIGURE 3-9. SAFETY SUBASSEMBLY.

The safety rod guide thimbles were locked to the lower reactor grid structure in a similar manner to that described for the control rod guide thimbles. Each safety rod was engaged to the common drive unit by a rotational locking mechanism. A hexagonal shaped collar on the upper end of the safety rod prevented inadvertent disengagement of the safety rod. This collar normally engaged the inside of the thimble, preventing rotation of the safety rod. To connect or disconnect the safety rod for loading purposes, the safety rod was raised 1 inch above its normal up position by the safety rod drive mechanism to raise the hexagonal collar above the thimble.

The safety rod upper adapter was identical to the control rod and the subassemblies, and was handled in the normal manner by the fuel transfer system. The guide thimble was removable in the same manner as the guide thimble for the control subassemblies.

Cooling of the safety rod was accomplished in the same manner as the control rod, but since it was a one-position device, no provisions were made for variable flow. The safety rods had to be in an up, most reactive, position before the reactor could be made critical or before fuel handling operations could be performed. It should be noted here, however, that this system was never called upon to perform its intended function, which was to shut the reactor down from an unintended reactor critical condition during reactor refueling and with the normal reactor control system inoperative. This feature could be characterized as an ultra-conservative feature that was not necessary and probably could be omitted in future liquid metal cooled fast breeder reactor designs.

Neutron source rods were placed in the outer blanket region. The antimony/beryllium combination provided the neutron source for neutron detector calibration when the reactor was shut down.

REACTOR VESSEL ASSEMBLY

The reactor vessel assembly (Figure 3-4 and Figure 3-10) consisted of the reactor vessel, the grid assembly, and the top cover. It contained the reactor-fuel and blanket subassemblies, and control and safety rods, and provided the proper configuration of these units. The assembly was located and supported at the bottom and on the centerline of the primary tank. It was supported on the structural members that reinforce the bottom of the primary tank inner shell. The vessel assembly was surrounded on all sides by the neutron shield and was submerged beneath approximately 10 feet of sodium.

The vessel assembly consisted of three major units: the grid-plenum assembly, the vessel, and the top cover. To ensure accurate alignment, the vessel was fastened to the grid-plenum assembly by bolts, which were tack-welded to ensure a permanent connection. The vessel cover served as a neutron shield as well as a closure. It was clamped to the vessel flange by means of three

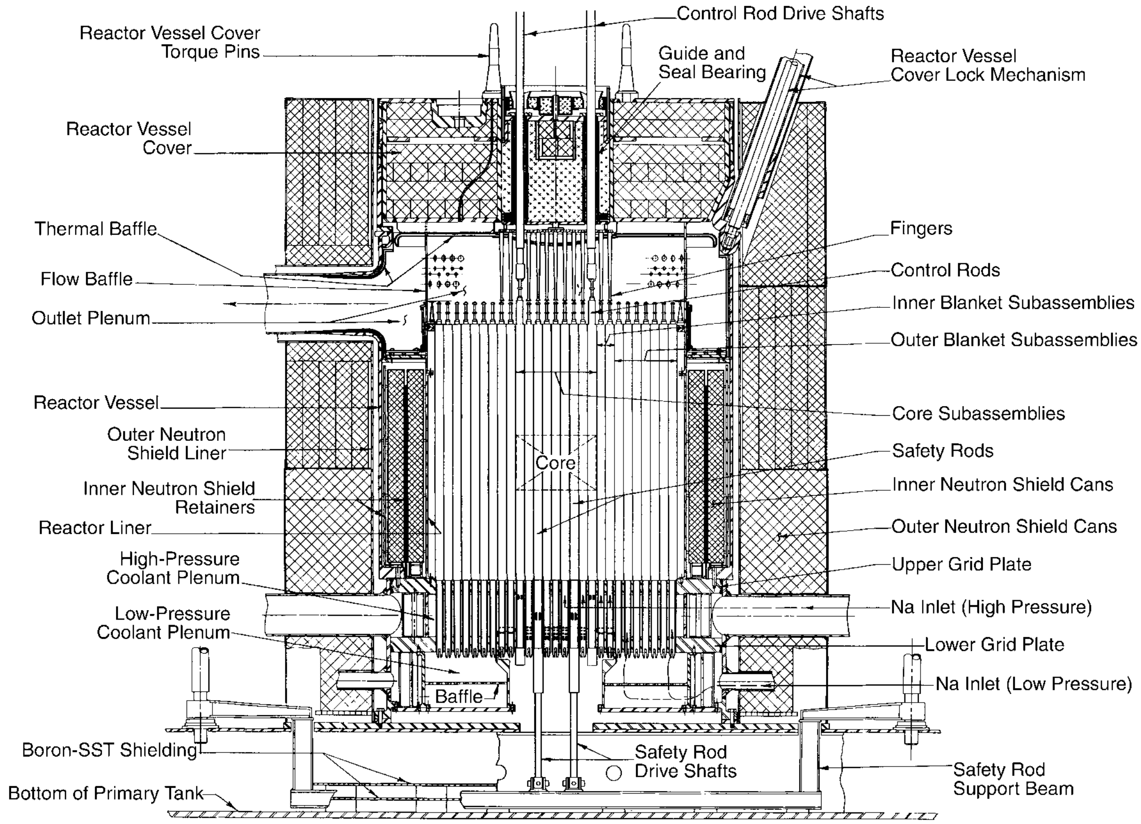


FIGURE 3-10. REACTOR VESSEL ASSEMBLY.

hold-down clamps (Figure 3-10 and Figure 3-11). When the cover was closed, it formed the upper reactor coolant plenum chamber from which the coolant flowed to the heat exchanger. Within the plenum chamber the coolant was at an average temperature of 900°F and a pressure of 18 pounds per square inch gauge. The cover separated this sodium from the ambient bulk sodium in the tank. The sodium seal was formed between the vessel flange and the cover, but some leakage occurred. When it was desired to exchange subassemblies, the hold-down clamps were released and the cover was elevated to allow the fuel handling system to unload the fuel below the raised cover, and transfer the fuel to the storage rack (Figure 2-8).

The reactor vessel was a cylindrical tank with flanged ends. The upper plenum of the vessel, as well as the coolant nozzle, was lined with a thermal baffle (Figure 3-10). The function of this baffle was to reduce the temperature difference across the vessel wall and also the coolant outlet nozzle wall. Below the plenum region the vessel contained a laminated steel thermal shield. The vessel wall was insulated from the bulk sodium in which it was submerged by a steel shell liner that was vented, and therefore contained static sodium. This shell and static sodium combination provided sufficient thermal insulation with acceptable thermal stresses in the vessel wall. The heat loss between the reactor outlet sodium and bulk sodium in the primary tank was relatively small and was not lost from the system.

The grid-plenum assembly (Figure 2-19 and Figure 3-5) was a combination structure that incorporated a grid to support and locate the subassemblies, and incorporated the coolant inlet plenum chambers that supplied coolant to the subassemblies. It consisted of two 4-inch-thick stainless steel plates that contained the locating holes for the lower adapters of the subassemblies. The subassemblies were supported by the upper plate and the lower adapters extended through the lower plate. A spherical shoulder on the

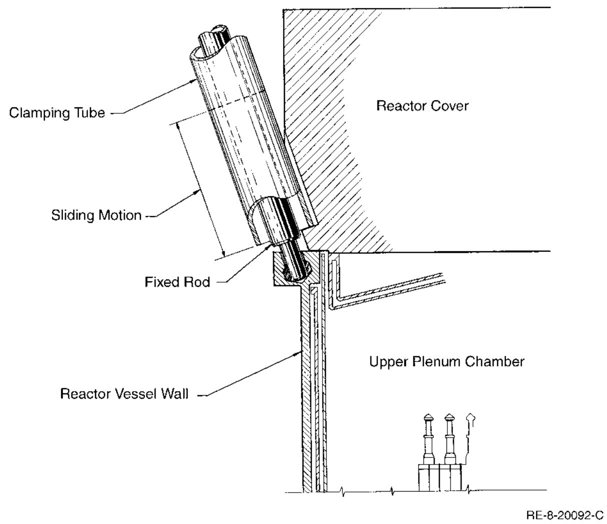


FIGURE 3-11. REACTOR COVER HOLD-DOWN.

subassembly that engaged a conical seat in the upper grid plate supported the subassemblies. This arrangement minimized the leakage flow of coolant along the outside surfaces of the subassemblies.

The high-pressure coolant plenum chamber supply for the core and inner blanket was formed between the two grid plates. The low-pressure coolant plenum chamber supply for the outer blanket consisted of an annular chamber immediately below the lower grid plate. Tubes welded to each plate in the outer blanket zone interconnected the upper and lower grid plates. This prevented short-circuiting of the high-pressure coolant through the outer blanket.

The grid-plenum assembly tube structure also provided the structural system required to support the entire reactor load on the upper grid plate. The high-pressure coolant flowed between these tubes into the core and inner blanket region where it entered the subassemblies. The lower nozzles of the core and inner blanket subassemblies contained holes located precisely between the upper and lower grid plates. The coolant entered the subassembly through these holes and flowed upward through the subassembly. The upper surface of the lower grid plate was stepped to close specific holes; this varied the cross-sectional area of the effective holes in the subassemblies. This arrangement provided orificing of the flow through the subassemblies to match the heat generation rate in each row of subassemblies as described earlier.

The lower end of the subassembly nozzles was closed, forming a hydraulic piston. The sodium in the high-pressure coolant plenum chamber was at a nominal pressure of 61 pounds per square inch, of which 8 pounds per square inch was static head (due to the sodium level in the primary tank). The remainder gave a pressure difference of 53 pounds per square inch acting across the piston. This provided a downward force, or hydraulic hold-down, of 148 pounds on the core subassemblies and 116 pounds on the inner blanket subassemblies.

The low-pressure coolant entered the low-pressure plenum chamber at 22 pounds per square inch, and entered the lower nozzles of the outer blanket subassemblies through openings at the bottom. Because the pressure drop through the outer blanket subassemblies was much smaller and the weight of these units was large, it was unnecessary to provide hydraulic hold-down.

Three different hole diameters for subassemblies were provided in the grid plate. This prevented a fuel subassembly from being inadvertently placed in the wrong position. To prevent the interchange of subassemblies in the other direction, subassembly angular orientation bars were used to provide proper angular orientation of the subassemblies in the reactor. They were fastened to the underside of the lower grid plate and engaged slots in the subassemblies. There were three thicknesses of bars: the core subassemblies engaged the thickest, the inner blanket subassembly slots were thinner and the outer blanket subassembly slots were the thinnest. If an inner blanket subassembly was inadvertently placed in a fuel position, the slot in the inner blanket subassembly tip was too narrow to engage the bar. This prevented engagement of the subassembly at least 2 inches short of its normal position in the grid, which was easily detected by the fuel handling mechanism. The same condition existed if an outer blanket subassembly was placed in an inner blanket position or a fuel position.

This method of loading control was adopted because a core subassembly inserted in either blanket zone introduced both a reactivity problem and a cooling problem, while a blanket subassembly introduced in the wrong zone introduced only a cooling problem. The lower grid was 19 inches deep, while the core was only 14 inches long. Since a core subassembly could not enter the grid in the wrong location because the diameter of the subassembly lower adapter was too large, a loading error would not permit the fuel section of the subassembly to enter the core region of the reactor. In the reverse manner, a subassembly could enter the grid for approximately 17 inches of travel, but the error was detectable.

The reactor cover provided the closure of the top of the reactor vessel and formed the upper surface of the outlet plenum chamber. It also provided the upper portion of the neutron shield. The 12 control rod drive shafts operated through guide, sleeves provided in the cover for these units. The fuel handling gripper mechanism and hold-down shafts also penetrated the cover. A small amount of leakage occurred through these various openings during reactor operation when a sodium pressure differential of approximately 12 pounds per square inch existed across the cover. This leakage flow was employed as a part of the neutron shield cooling system in this region. This too represented a small heat loss because this hot sodium bypassed the IHX but this heat was not lost from the system; it was recycled through the reactor.

The top cover was raised and lowered by two shafts penetrating the small rotating plug. The cover was fastened to the reactor vessel by three clamping mechanisms, and the raising and lowering mechanism was designed to permit free expansion of the two lifting shafts. The drive shafts for the three clamping mechanisms were also permitted to float. This arrangement avoided the large load due to internal pressure being transferred to the cover lifting mechanism, and also avoided problems associated with differential thermal expansion in the system.

The underside of the reactor cover had projections on the same spacing as the core and inner blanket subassemblies. These pins were positioned directly above each subassembly adapter and provided approximately 1/4 inch of clearance between the adapter and the end of the pin. The pins prevented any appreciable lifting of the subassemblies in the event of failure of the hydraulic hold-down system or vertical movement for any reason.

Thermocouple wells were provided adjacent to some of these pins to measure the outlet sodium temperature in various regions of the reactor. The thermocouple leads were introduced through tubes that were brought out through the hollow cover lifting drive shafts. Inside the cover the tubes were routed to the various locations. The tubes were permanently installed in the cover, but the thermocouple junctions and leads could be replaced.

PRIMARY COOLING SYSTEM

The primary system component arrangement is shown in Figure 2-11. The reactor vessel was centrally located at the bottom of the primary tank. The pumps, heat exchanger, and connecting piping were disposed radially around the reactor vessel and elevated above it.

The coolant flow path in the primary cooling system was as follows:

- Two primary coolant pumps took suction from the bulk sodium in the primary tank.
- The flow from each pump separated into two streams before entering the high-pressure and low-pressure reactor inlet plenum nozzles.
- The 12-inch inlet nozzles to each of the high-pressure plenums were approximately diametrically opposite each other.
- Each pump outlet was connected to the corresponding high-pressure inlet plenum nozzle.
- A smaller line connected to each outlet line provided a take off flow through a flow control valve to each low-pressure plenum nozzle.

These valves were set during initial plant operation and remained fixed during most of the operating lifetime of the plant.

Coolant flow in all regions of the reactor was upward through the fuel and blanket subassemblies and into a common upper plenum chamber with a single 14-inch outlet. The heated sodium flowed to the shell side of the intermediate heat exchanger through a permanently installed 14-inch pipe. The pipe had a Z configuration to accommodate thermal expansion. The auxiliary pump was installed in the upper horizontal leg of this outlet pipe line.

The primary coolant flowed downward through the shell side of the heat exchanger and discharged into the bulk sodium in the primary tank (as shown in Figure 2-11). The heat exchanger primary sodium outlet was approximately 7-1/2 feet above the centerline of the reactor. This arrangement assured an inherently reliable natural convection cooling system for shutdown cooling without heat removal by the secondary sodium as discussed earlier.

The sodium line between the upper plenum of the reactor vessel and the heat exchanger shell was permanently attached to these two components. The heat exchanger shell was permanently attached to the underside of the cover of the primary tank; however, the tube bundle, including the upper and lower secondary sodium plenums, secondary sodium inlet and outlet nozzles, and shield plug (as shown in Figure 3-12) could be removed as a unit in a vertical direction.

When the reactor was in operation, coolant was supplied in parallel by the two main primary sodium pumps operating. At 100 percent power operation, each pump supplied approximately 4,700 gallons per minute of coolant at 55 pounds per square inch head; the maximum capacity of each pump was approximately 5,000 gallons per minute at 85 pounds per square inch .

The two primary sodium pumps were vertical-mounted, single-stage, centrifugal-type mechanical pumps (Figure 3-13). These pumps employed a gas-tight motor and sodium hydrostatic bearing. It should be emphasized that the success of this bearing made the use of mechanical centrifugal pumps possible.

Variable-speed motors powered by a motor-generator set providing variable voltage and frequency drove these pumps. They were controllable from about 20 to 100 percent speed with specified rates of acceleration and deceleration. The direct coupled pump drives were special, totally enclosed, gas-tight, 480-volt, alternating current motors. Motor cooling was provided by forced circulation of air through an air-to-argon gas heat exchanger within the sealed motor enclosure as shown in Figure 3-13. Labyrinth-type shaft seals were employed to minimize diffusion of sodium vapor into the motor enclosure.

The inlets to the pumps were open to the primary bulk sodium in which they were submerged. The entire volume of sodium—approximately 86,000 gallons—was at the reactor inlet temperature of approximately 700°F. It was heated approximately 200°F as it passed through the reactor and was cooled approximately 200°F as it passed through the intermediate heat exchanger. From there it returned to the bulk sodium at about 700°F; when the reactor was at power, this was the primary sodium temperature scenario. At full power, about 62.5 megawatt thermal were generated and transferred in this manner. It was achieved by forced circulation of sodium through a very simple heat generation and exchange thermal system. The two primary pumps provided the forced circulation of the primary sodium coolant to achieve this capability under all of the power conditions at which the reactor operated.

This physical arrangement of the primary system components simplified the system enormously. A traditional system of pumps, reactor, and heat exchanger would involve much more piping and support facilities. On the other hand, access to these components submerged in high temperature sodium complicated service and maintenance.

As described earlier, the intermediate heat exchanger internals were completely removable for maintenance or even replacement. Similar capability was required for the pumps. Ball seat type disconnects were used between the pump outlet nozzle and the permanent piping to the reactor inlet plenum. This allowed for removal of these pumps from the primary tank as shown in Figure 3-14. During the operating lifetime of the EBR-II, each pump was removed only two times. Figure 3-15 is a photo of a pump after removal from the caisson used for removal and cooling.

SHUTDOWN COOLING SYSTEM

When the reactor was not operating, fission product decay heat had to be removed. The systems that removed the heat generated at power were perfectly capable of performing the same function when the reactor was shut down. However, these systems were not sufficiently dependable to meet the reliability requirement involved. Shutdown cooling was required at all times and had to be absolutely reliable. This unique requirement of nuclear reactors was particularly important in fast reactors because they operate at very high power density.

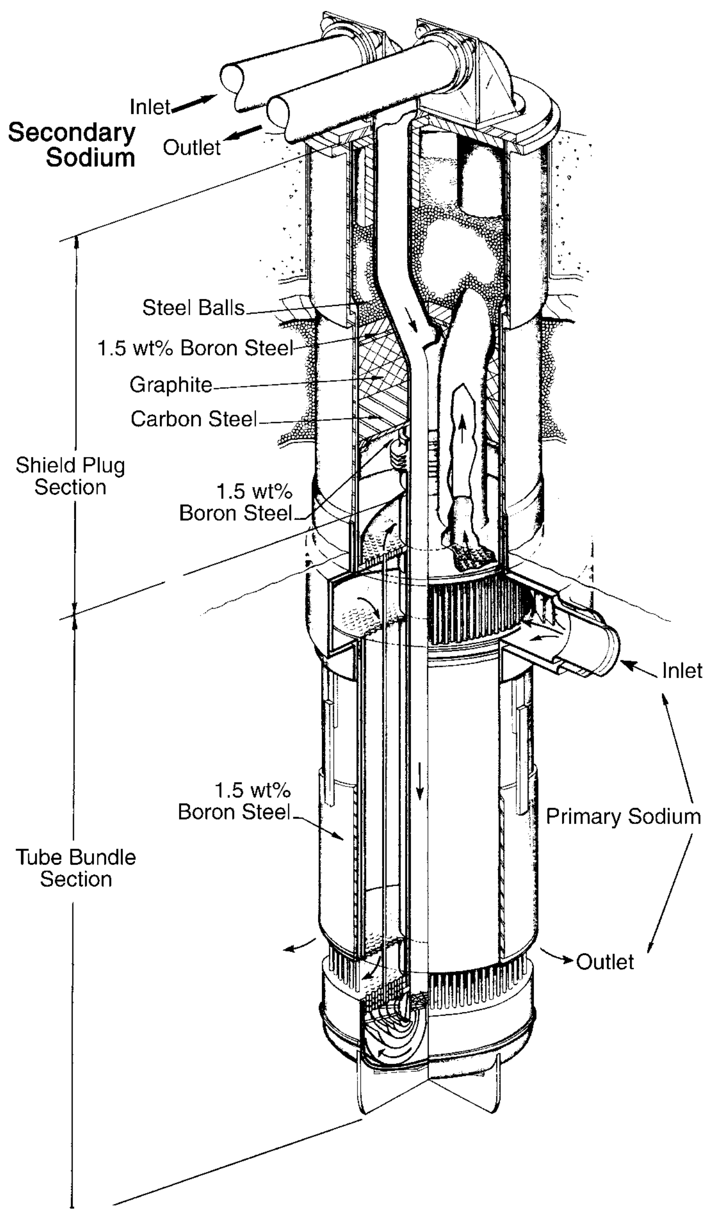


FIGURE 3-12. HEAT EXCHANGER.

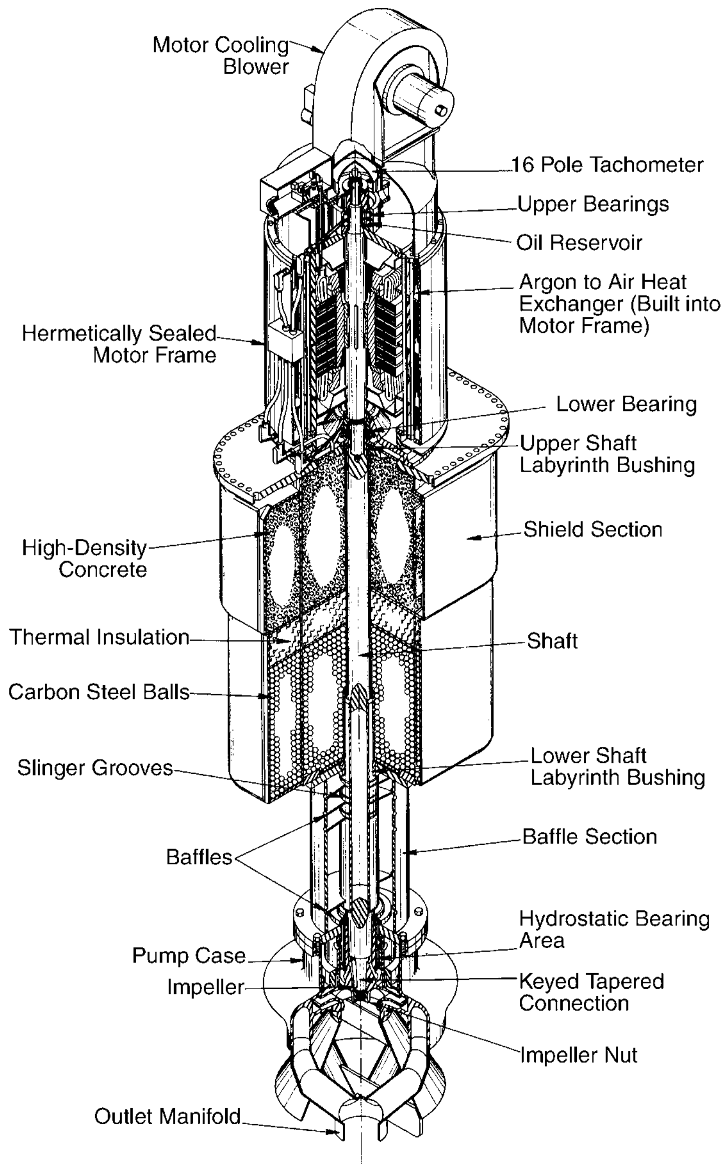


FIGURE 3-13. EBR-II PRIMARY PUMP.

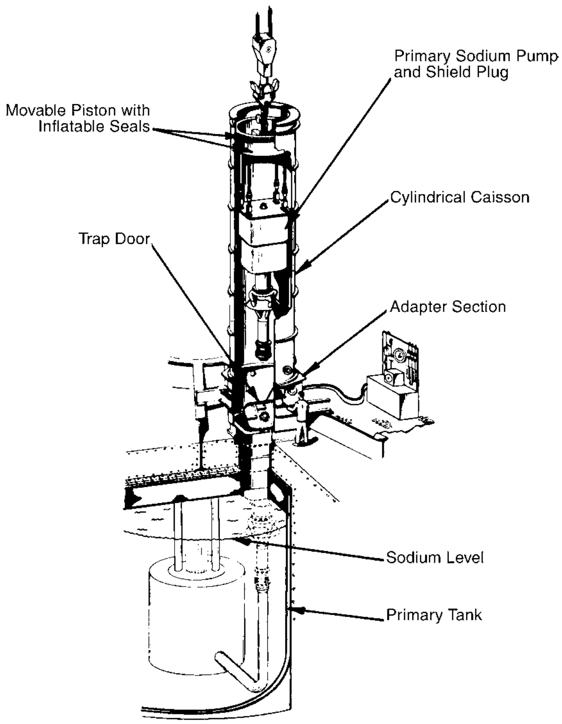


FIGURE 3-14. EBR-II PRIMARY PUMP REMOVAL.

As noted earlier, the EBR-II concept incorporated totally passive systems to remove fission product decay heat from the fuel. The natural thermal convection of sodium through the reactor was enhanced by a unique auxiliary sodium pumping system. This system augmented thermal convection as needed under certain conditions of reactor shutdown which could have inhibited the transition from forced convection coolant flow to natural convection coolant flow.

The auxiliary pump ensured continuity of flow under these conditions and prevented undesirable temperature transients. The auxiliary pump was a direct current electromagnetic pump located in the reactor outlet line, and operated in series with the main pumps. Its design capacity was approximately 500 gallons per minute at 0.15 pounds per square inch and 900°F sodium temperature. The pumping section was incorporated in the 14-inch outlet pipe, with no change in pipe cross section. This was done to maintain the integrity of the piping system at the expense of pumping efficiency, which was not important.

The auxiliary pump electrical power was supplied from metallic rectifier units and storage batteries. The storage batteries, operating in parallel with the rectifier units, assured pump operation in the event of a complete power failure. During normal operation, these batteries floated on the line and remained fully charged at all times. In the event of a sustained power failure, the pump operated until the battery was discharged, which resulted in a gradual decay of the flow rate and an ideal transition to thermal convection.

Removing the fission product decay heat from the reactor fuel after shutdown involved heat removal from the reactor by the primary sodium flowing through the reactor; and heat removal from the primary sodium. After reactor shutdown, coolant flow through the reactor was maintained as follows:

- Operation of the main pumps
- Operation of the auxiliary pump
- Natural convection flow.

Heat removal from the sodium leaving the reactor could be accomplished by two methods:

- The heat could be transferred to the secondary system, then to the steam system, and eventually to the atmosphere.
- The heat could be transferred to the bulk sodium in the primary tank and then transferred to the atmosphere more directly.

If the reactor cover was closed, coolant flow through the reactor by any of the three methods described above followed the normal circuit through the heat exchanger to the bulk sodium. If the secondary system was operating, the heat was transferred in the heat exchanger to the secondary system sodium. The secondary system, in turn, transferred heat to the steam system in the steam generator. The heat left the steam system via the condenser, and was transferred to the atmosphere through the cooling tower.

If the secondary system was inoperative, the heat was transferred to the bulk sodium in the primary tank. The heated sodium leaving the reactor was mixed with the bulk sodium by discharging from either the heat exchanger, or, if the reactor vessel cover was raised, from the top of the reactor. The heat was then removed from the bulk sodium by the shutdown coolers that, in turn, transferred the heat to the atmosphere through a finned-tube air

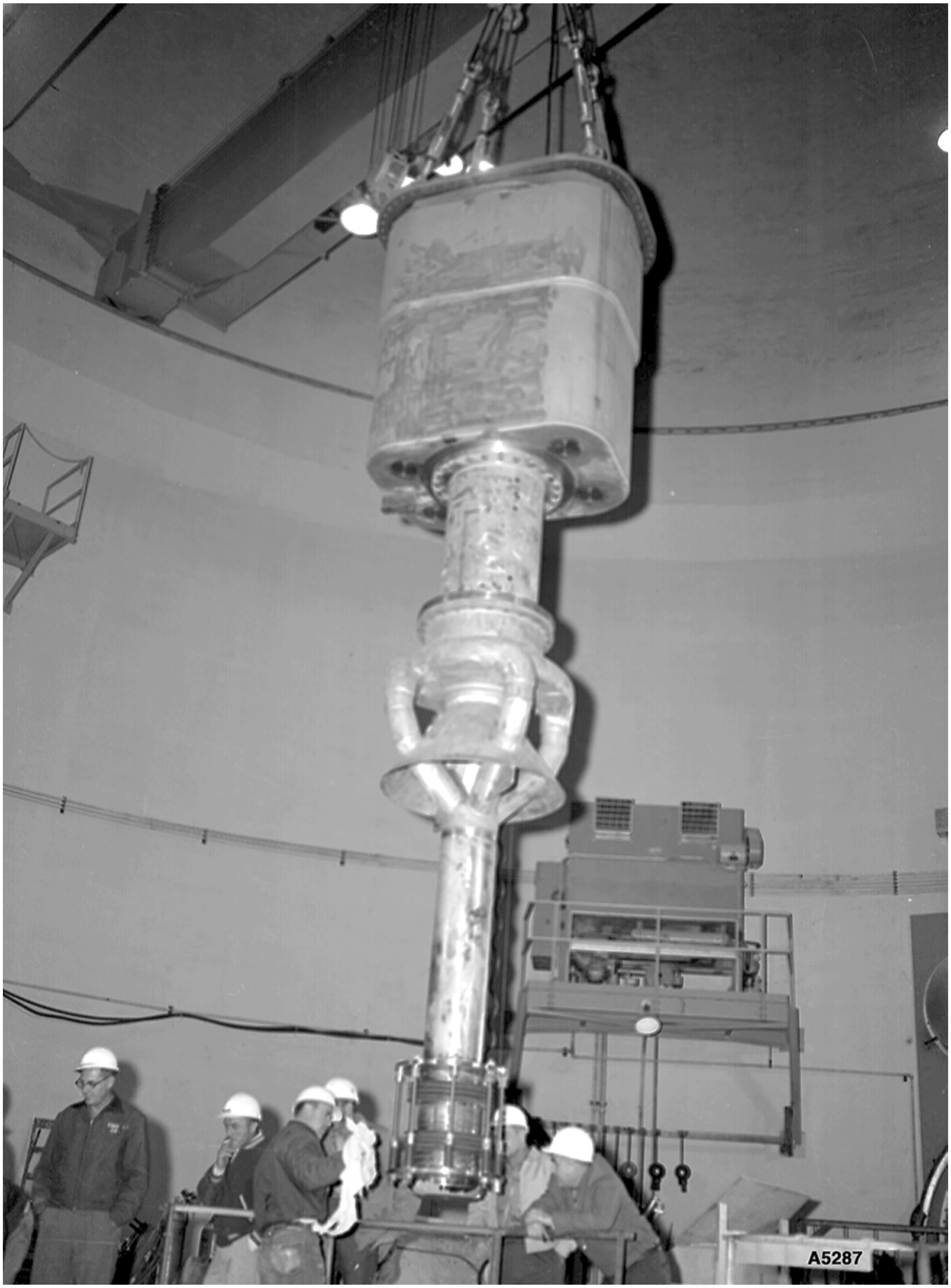


FIGURE 3-15. PRIMARY PUMP AFTER REMOVAL.

heat exchanger. Since the primary system had a very large thermal capacity compared to the amount of fission product decay heat removed from the reactor, the temperature rise of the bulk sodium was slow, and fast response of the shutdown coolers was not necessary. The salient feature of this method of heat removal was the complete independence from any external power source. All fluid flow was due to natural convection.

The shutdown cooler (Figure 3-16) was an immersion-type bayonet heat exchanger. Basically, it consisted of two concentric pipes approximately 26 feet long, the outer pipe being closed at the bottom. An inner concentric pipe produced an annulus between the two concentric pipes. Coolant flowed down through the central pipe and up through the annulus as shown. To enhance thermal convection, the inner pipe was insulated by a void space between two concentric pipes to minimize heat transfer to the downward flowing coolant and thus enhanced heat transfer and thermal convection in the annulus. The coolant was sodium-potassium eutectic that was liquid at room temperature. The bayonet heat exchanger was installed in a thimble with a static sodium bond between to provide effective heat transfer from the bulk sodium surrounding the thimble. This very conservative arrangement incorporating an extra thimble was provided to avoid contamination of the primary sodium with potassium in the event of a coolant leak in the bayonet cooler.

The coolant entered the inner pipe of the bayonet cooler at the top and flowed downward to the bottom of the inner pipe where it reversed direction and entered the annulus. The flow was then upward through the annulus, where heat transfer to the coolant occurred. Leaving the bayonet cooler, flow was upward into a finned-tube air heat exchanger, which was located in a dampered air stack outside the reactor containment building. Here the heat was transferred to the atmosphere by natural convection of air; the cooled sodium-potassium eutectic then flowed downward into the inlet of the bayonet cooler. The balance of the system is shown schematically in Figure 2-13.

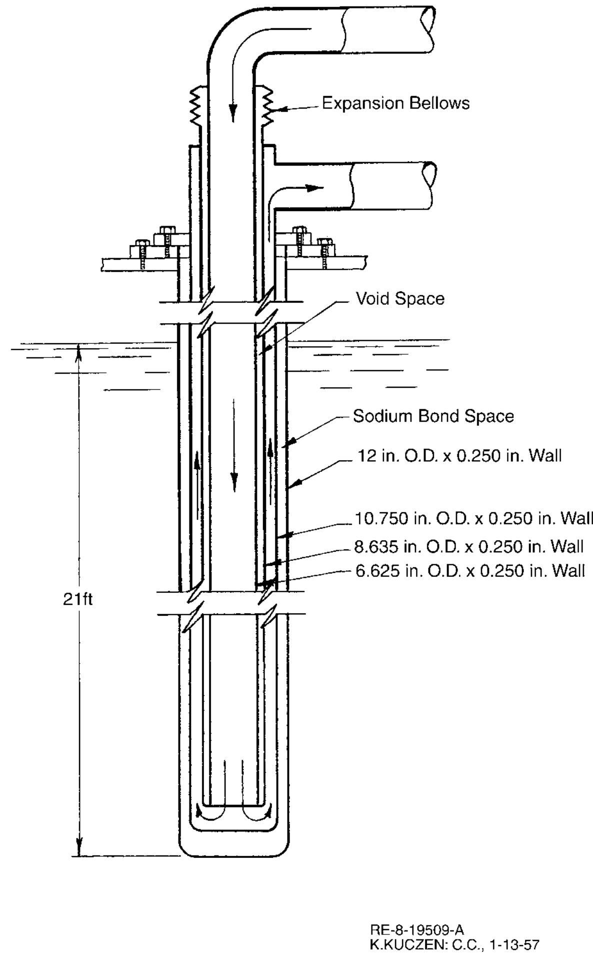


FIGURE 3-16. SHUTDOWN COOLER.

The rate of heat release from the system was controlled by the position of the stack dampers. Normally the dampers were actuated by automatic control, however manual control was also incorporated in the event of failure of the automatic system. During reactor operation, the dampers were held closed by electrically energized magnets, and a minimum flow of sodium-potassium eutectic occurred in the shutdown cooling system. This method of operation prevented the freezing of the coolant in cold weather, provided for positive starting when the dampers were opened, and also reduced thermal shock on the system. When the stack dampers were opened the thermal head on both the coolant and air side was increased. This gave rise to increased flow of both fluids which in turn, resulted in increased heat removal from the bulk sodium.

The sodium-potassium eutectic cooling system, external to the bayonet cooler, was instrumented with thermocouples and an electromagnetic flow-meter. An alarm system was interlocked with these measuring devices to annunciate and indicate abnormal conditions of flow or temperature.

The system was designed for maximum reliability and simplicity. The design of the bayonet coolers provided for minimum internal stresses over large temperature ranges and minimum obstructions in the flow circuit. All welded construction was used and no valves were incorporated in the system.

NEUTRON SHIELD

The neutron shield surrounded the outside of the reactor vessel on all sides and was submerged in the bulk sodium of the primary tank. The shielding material was graphite and graphite impregnated with 3 percent (by weight) of natural boron. To prevent the reaction and contamination of the graphite with sodium, it was canned in stainless steel.

For purposes of description, the shield could be separated into three sections: radial, top, and bottom as shown in Figure 3-17. To facilitate fabrication, handling, and installation, the graphite and the borated graphite were canned in conveniently sized pieces that could be readily stacked and placed in position around the reactor vessel. All cans used for cladding were leak tested, loaded with graphite, and closed by welding. The 1/8-inch clearance space between the can and the graphite was filled with helium. The cans were filled with helium to an absolute pressure of 10 inches mercury, at room temperature, to minimize the internal pressure at operating temperature (12 pounds per square inch absolute at 700°F) and also to provide a heat transfer medium to conduct the heat generated in the graphite to the can wall. The helium generated by the (n,α) reaction with boron, was expected to result in an increase in pressure of approximately 19 pounds per square inch (at operating temperature) during the life of the reactor. This assumed that all of the helium generated in the graphite would be released to the helium atmosphere in the stainless steel can. The cans were designed for a positive internal pressure 50 pounds per square inch greater than the external pressure. They were cooled externally by natural convection flow of sodium.

The radial shield was assembled from graphite blocks fitted in stainless steel cans stacked in three levels to a height of approximately 13 feet. Two rows of canned graphite were positioned inside the reactor vessel periphery and five rows around the periphery of the reactor vessel. Each row was held in place by stainless wire mesh. Clearance was provided between the cans to permit natural convection flow of sodium. Each row was staggered with respect to adjacent rows to minimize neutron streaming. Specially shaped shielding cans were used around the inlet and outlet sodium pipes of the reactor vessel and around the instrument thimbles that terminated in the neutron shield. Retainers and liners provided positive positioning of the shield cans and enhanced natural convection cooling of the shield.

Because of the complex structure of the reactor vessel cover, the cans in the cover were of complex shape. They were stacked to prevent neutron streaming and to permit cooling. The cover contained six layers of cans filled with either

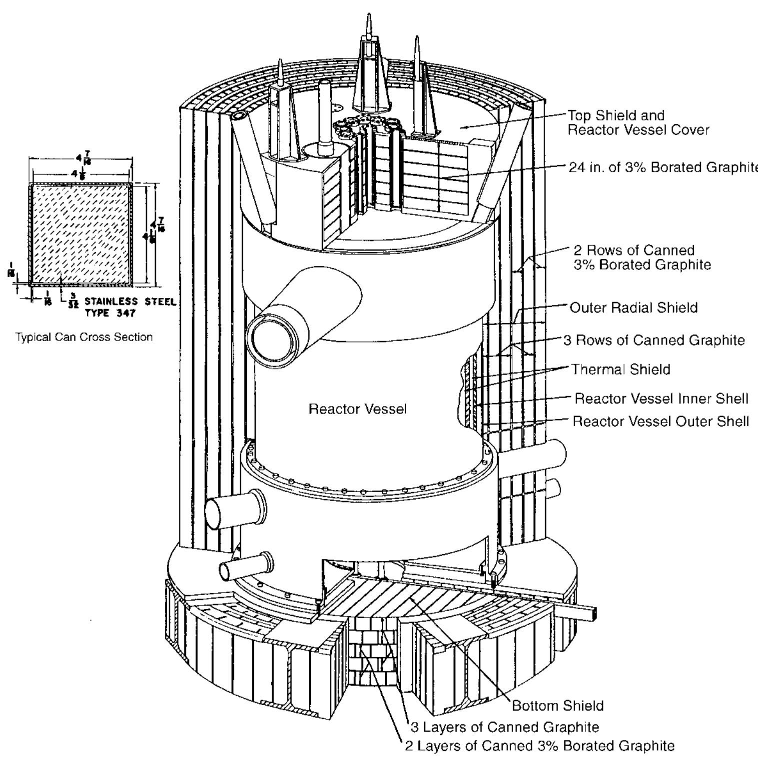


FIGURE 3-17. NEUTRON SHIELD.

3 percent borated graphite or boron carbide. The total thickness of the top shield was 24-3/4 inches.

The bottom shield consisted of borated stainless steel plates located between the vertical webs of the beams on the bottom of the primary tank. The heat generation in the structure below the reactor was not critical. This arrangement provided adequate shielding and a simple structure below the reactor vessel.

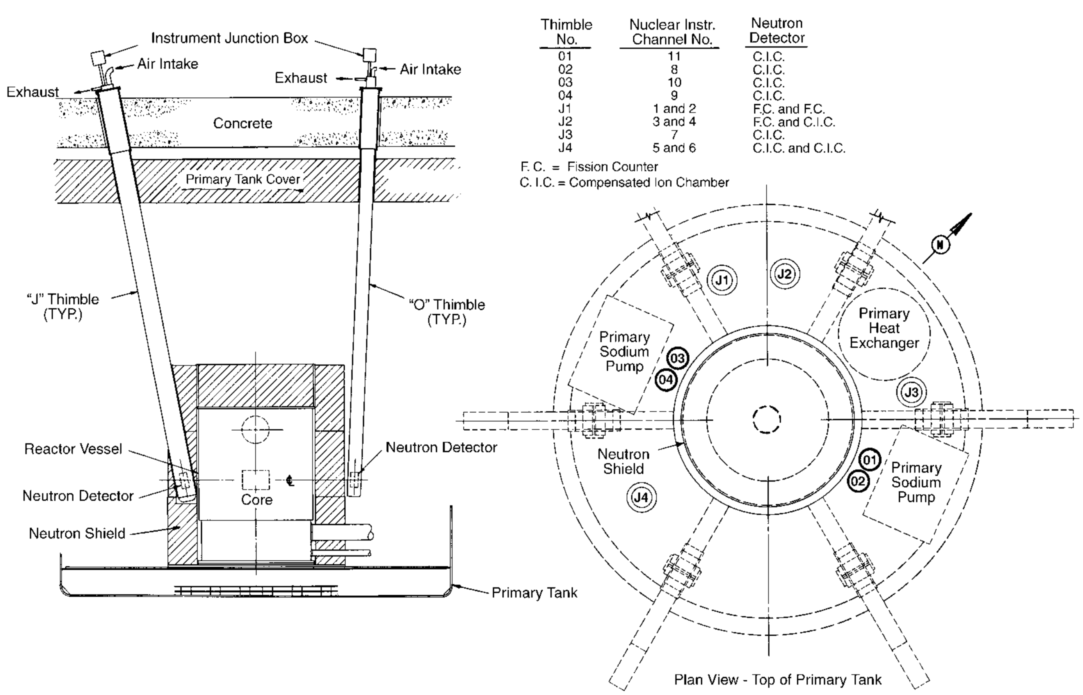
COUNTERS, CHAMBERS, AND INSTRUMENT THIMBLES

Three fission counters and eight compensated ion chambers comprised the detectors for the nuclear instrument channels. Since detectors of proven reliability for 700°F operation were not available, conventional detectors were employed in air cooled thimbles. For reliable operation, the temperatures of counters and chambers were maintained below 140°F.

Three uranium-235-enriched fission counters detected thermal neutrons in the startup range of operation. These counters were positioned in "J" thimbles located in the radial neutron shield. Eight compensated ion chambers of the boron-coated type were located adjacent to, or in the radial neutron shield.

The general arrangement of the nuclear instrument thimbles and their associated fission counters and ion chamber is shown in Figure 3-18. Eight thimbles were provided. Four "J" thimbles were imbedded in the radial graphite neutron shield outside of the reactor vessel, and four "O" thimbles were located immediately adjacent to the neutron shield.

Thimble cooling was accomplished by drawing room air through the thimbles. The system did not recirculate, the exhaust air was combined with the biological shield cooling air and discharged through the Fuel Cycle Facility 200-foot stack. Two full capacity blowers were available, one was on standby. For maximum reliability, the standby blower system was provided with automatic switch-over to a 100 kilowatt auxiliary diesel generator power supply. This was in addition to the 400 kilowatt plant auxiliary diesel generator. The reactor was scrammed in the event of thimble cooling failure, but these additional precautionary measures were designed to protect the nuclear detectors and pre-amplifiers from thermal damage. It should be noted that the primary function of these diesel generator power supplies was to protect against economic loss, not reactor damage.



Thimble No.	Nuclear Instr. Channel No.	Neutron Detector
01	11	C.I.C.
02	8	C.I.C.
03	10	C.I.C.
04	9	C.I.C.
J1	1 and 2	F.C. and F.C.
J2	3 and 4	F.C. and C.I.C.
J3	7	C.I.C.
J4	5 and 6	C.I.C. and C.I.C.

FIGURE 3-18. LOCATION OF NUCLEAR INSTRUMENT THIMBLES.

CONTROL AND SAFETY DRIVE SYSTEMS

Twelve control rods controlled the operation of the reactor. Each rod was independently driven by an electrical-mechanical drive mechanism. The drives were identical and were so arranged that only one drive could be operated at a time, with the exception of scram when all 12 operated simultaneously. Operating control was achieved by a 14 inch vertical motion of the control rods that was provided by a rack and pinion-type drive with constant-speed electric motors, therefore, only one speed of movement was possible. The control rods were disconnected from their drives during fuel handling operations. The disconnect was made with the control rods in their down or least reactive position. The control rods remained in this position during fuel handling operations.

Two safety rods were provided in the reactor in addition to the 12 operational control rods. The safety rods were not a part of the normal operational control system of the reactor. The safety rods were always in the reactor and they were designed to function when the control rods were disconnected from their drives. The primary purpose of the safety rod was to provide available negative reactivity when the reactor was shut down and the control rods were disconnected. They provided a safety factor during reactor loading operations. The safety rod drive mechanisms were separated from and completely independent of the control drive systems. The drives and vertical drive shafts were located outside the reactor and rotating plugs, completely independent of the fuel handling system and reactor components as shown in Figure 3-19. Low level detectors separate from the normal operational control system actuated them.

The control rod drive mechanism performed three major functions: the connection between the drive and the control rod, the slow-speed vertical motion (in both directions) for reactor control, and the high-speed downward motion for reactor scram. These operating functions were combined in a single unit and were appropriately interlocked to ensure proper operation.

The control rod drive mechanism was attached to the control rod by means of a gripper. The gripper attached to the conical top of the control rod adapter (which was also used for fuel handling operations of the control rods). The gripper consisted of two jaws that engaged the control rod adapter and was operated by a cam incorporated in a sliding sleeve; the engagement was very similar to the fuel handling gripper to subassembly engagement during fuel handling operations. Jaw operation was positive; the jaws were opened and closed by the cam, and were locked in position by the cam. The jaws operated through a funnel-shaped guide tube and upon opening, receded beyond the guide tube, providing a smooth interior surface. This eliminated the possibility of the control rod adapter hanging up after the jaws were opened.

The gripper also contained a sensing device that made contact with the top of the control rod adapter. It consisted of a plunger made to move 1/2 inch in a vertical direction by the control rod adapter during engagement and disengagement of the control rod from the gripper. It was spring-loaded and the motion of the sensing plunger was transmitted to a position indicator. If necessary, it could also be used to forcibly eject the adapter from the gripper. A third check was also provided to eliminate the very unlikely possibility of the control rod adapter sticking to the sensing plunger. The relationship between the control rod adapter, the sensing plunger, and the gripper jaws was such that after the control rod was released, and the plunger was in the down position, the jaws would not close if the adapter was still in contact with the sensing plunger. Closing the jaws after the control rod had been released provided a final check that release had actually been accomplished. The arrangement of the units comprising the gripper mechanisms is shown in Figure 3-20.

The gripper device was attached to the main shaft, which extended upward through the biological shield into the operating area above the primary system. The actuating mechanism for the gripper and the sensing mechanism were located above the operating floor and were easily accessible for inspection and maintenance. The necessary motions employed to actuate the gripper and to sense the operations were transmitted by shafts from the gripper to the operating stations. The actuating mechanism shown in Figure 3-21 was constructed in such a way that the control rod could not be released except when it was in the down position of the control stroke. The position of the jaw actuating device and the position of the sensing device were indicated by transducers and were suitably interlocked into the system. The actuating device had to be in its proper position, and the sensing device had to affirm that it was, before subsequent operations could be performed.

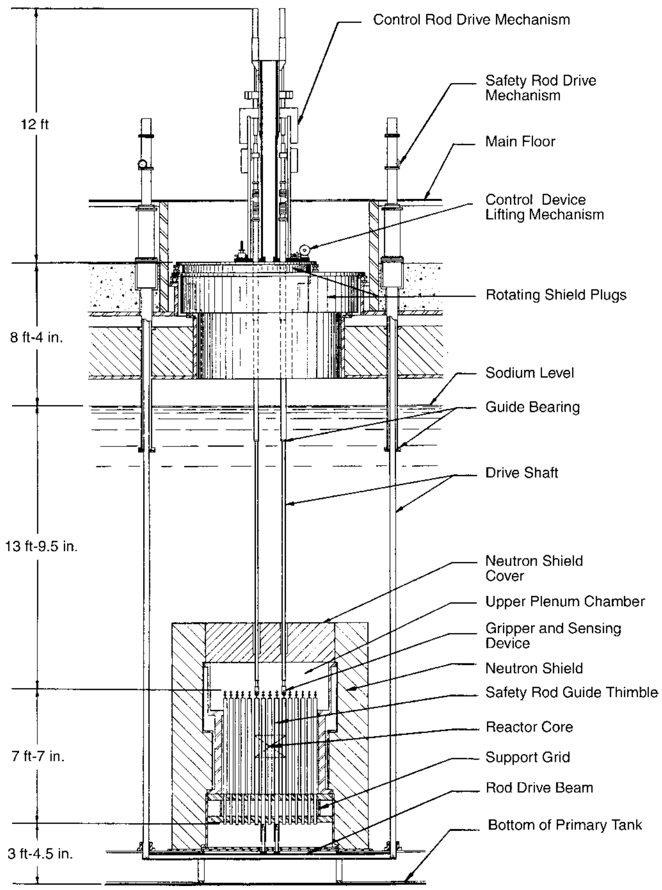


FIGURE 3-19. EBR-II CONTROL AND SAFETY ROD DRIVE SYSTEM.

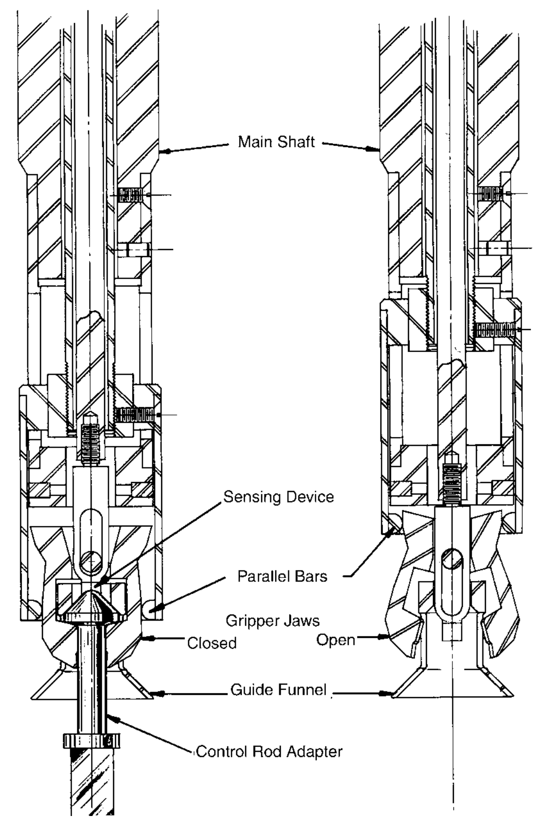


FIGURE 3-20. CONTROL ROD GRIPPER MECHANISM.

The control rod was actuated by a long shaft that extended through the upper biological shield with the control rod attached to its lower end and the drive mechanism at its upper end. The shaft was driven by a rack and pinion at a rate of 5 inches per minute by a constant-speed instantly reversible, polyphase motor. The rack gear teeth were cut on the outside of the tube through which the main drive shaft extended. The drive shaft was connected to the rack tube by a fast-acting magnetic latch. The latch consisted of two rollers that engaged notches in the shaft and were actuated by a magnetic clutch. The magnetic clutch was energized to engage the latch and thereby connect the shaft to the rack tube. The latch arrangement is shown in Figure 3-22.

The main shaft extended upward through the rack tube and was attached to a piston in a pneumatic cylinder. The upper end of the cylinder contained compressed air at a pressure of approximately 50 pounds per square inch gauge. The lower end of the cylinder was open to the atmosphere. The pneumatic pressure was always acting against the piston, tending to drive the shaft, and thus the control rod, down. The latch connecting the shaft to the drive rack prevented motion. Upon a scram signal, the magnetic clutch was de-energized, releasing the shaft from the drive rack and driving the control rod down, out of the reactor core. Scram could occur at any position in the operating stroke of the control rod and was automatically actuated by a power failure, which de-energized the magnetic clutch. This was accomplished in a release time of 0.008 second, including the time elapsed between actuating the scram signal and beginning of shaft motion. To ensure the compressed air supply to the air cylinder, accumulator tanks were provided, which in turn were supplied by an air compressor. Check valves were provided in the connecting lines between the accumulator tanks and the air cylinders, and between the air compressor and the accumulator tanks, to prevent loss of compressed air in the

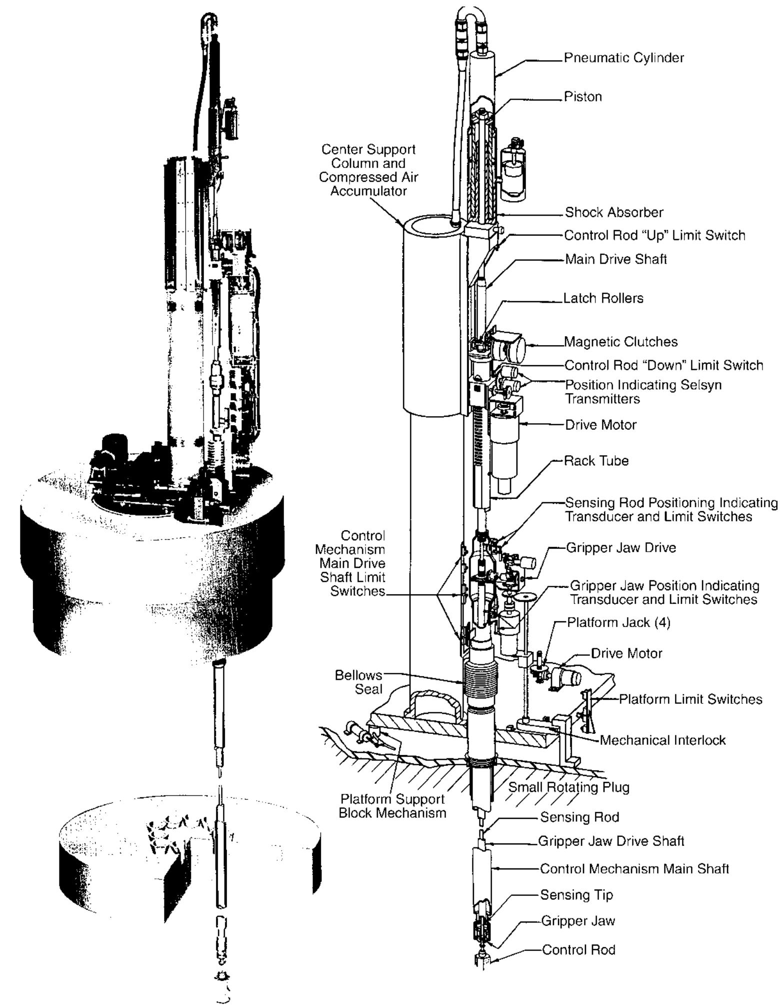


FIGURE 3-21. CONTROL ROD DRIVE MECHANISM.

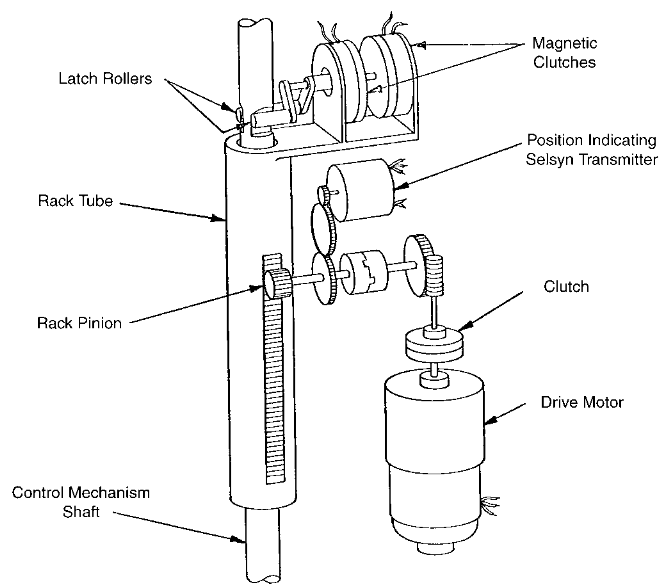


FIGURE 3-22. CONTROL DRIVE AND LATCH MECHANISM.

event of line failure. Pressure actuated switches scrammed the reactor in the event of failure of the air supply. The compressed air available in the cylinder or in the accumulator tanks was sufficient to insure pressure assist during a scram, in addition to the force of gravity. Deceleration of the scram stroke was accomplished by a hydraulic shock absorber connected to the air cylinder. The shock absorber was actuated during the lower 5 inches of travel.

A mechanical stop for upward motion, when the piston reached the top of the air cylinder was built into the system. If the limit switches on the rack driving pinion failed to stop the unit at the upper end of its travel, the drive shaft was stopped, including the control rod, and the rack continued to travel, moving away from the shaft and disengaging the latch. When this occurred, the shaft and the control rod were automatically scrammed by the disengagement of the latch. Over travel of the control rods was prevented and was not dependent upon the operation of the limit switches.

The 12 control drive mechanisms were mounted on a platform that surrounded a central support structure. The platform could be raised 3 inches and lowered 3/4 inches from its normal operation position. The upward movement was required to raise the lower end of the drive mechanisms, after disconnect from the control rods, to clear the subassembly adapters during fuel handling operations. The bottom position of the normal control rod stroke held the control rod 3/4 inches above its bottom seat in the guide sleeve. When released from the gripper, the control rod dropped and was supported by the control and thimble guide sleeve. The downward movement of the platform was required to engage the control rods and grip them when they were in the down position.

The design of the control rods and drive systems was extremely challenging. The space available was very limited due to the close spacing of the rods in the reactor and the demands for very high reliability of operation. A photo of a single drive unit and the cluster of 12 drives is shown in Figure 3-23.

The two safety rods (Figure 3-9 and Figure 3-10) were connected beneath the reactor to a horizontal bar which was connected to two vertical shafts which extended upward outside the biological shield. Each shaft was coupled to a rack tube by a magnetic clutch latch arrangement similar in design to one described above for the control rod drive. The rods were driven by synchronous motor drives and simply raised the system to the cocked position. When the latch was released, the drive mechanism and the safety rods fell 14 inches under the force of gravity. A pneumatic shock absorber decelerated the mechanism during the last 5 inches of movement. All reactor operations, including actuation of the control system or actuation of the fuel handling system, required the safety rods to be in the up position. The safety rods were connected to the horizontal support bar and the entire system acted as a unit with the support bar and both rods being dropped simultaneously.

FUEL HANDLING SYSTEM

EBR-II utilized a series of unique processes to handle reactor fuel (and blanket and other reactor components). These processes were divided into two broad categories:

- Those that were performed with the reactor shut down, designated as unrestricted operations.
- Those that were performed with the reactor in operation, designated as restricted operations.

The restricted operations could be further subdivided into fuel transfer operation and fuel transport operation.

Fuel handling operation involved the movement of subassemblies between the reactor and intermediate storage in the storage rack. All of these operations were performed in the primary tank with the subassemblies submerged in, and cooled by, sodium. These operations were performed with the reactor shutdown, and the control rods disconnected from their drives. Since these operations were performed with the subassemblies submerged in sodium they were not visible. Although the process consisted of a series of relatively simple operations, they were complicated by lack of visibility. They are described here in considerable detail to convey the level of attention that was provided to achieve reliable operation.

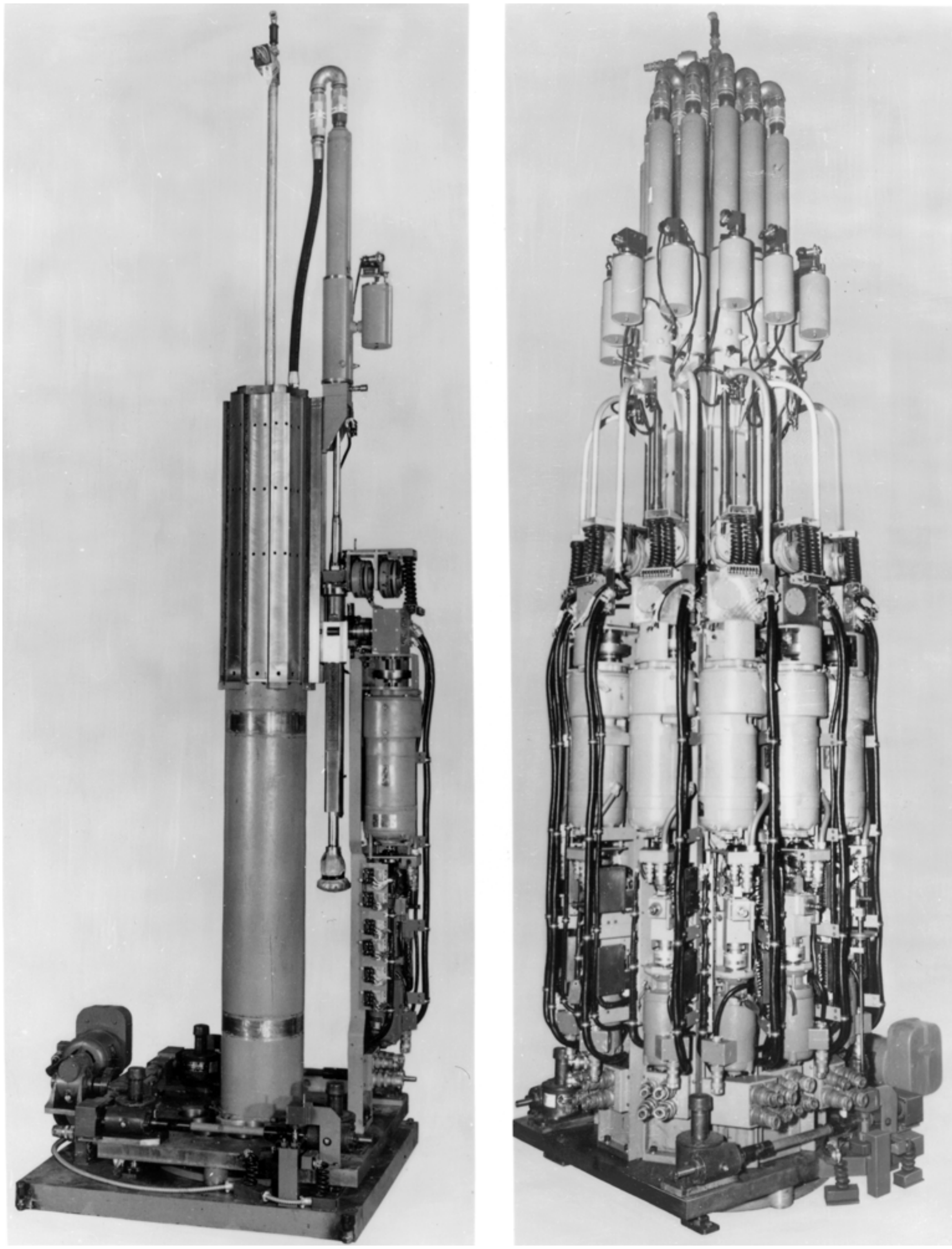


FIGURE 3-23. SINGLE DRIVE UNIT AND CLUSTER OF 12 DRIVES.

Fuel transfer operations involved the transfer of subassemblies between the storage rack and the inter-building coffin. This operation involved the use of the fuel unloading machine to effect this transfer, including the transition from sodium as the coolant to gas as the coolant for spent assemblies and the transition from gas to sodium for new or reprocessed subassemblies. These transitions occurred as the subassemblies were transferred between the sodium environment in the primary tank and the inert gas environment in the fuel unloading machine and vice versa. Fuel transport operations involved the transport of subassemblies between the inter-building coffin transfer station in the Reactor Plant and the air cell in the Fuel Cycle Facility. Included were the transport through the equipment air lock between the buildings, which maintained the containment integrity of the reactor containment building. Both the fuel transfer and fuel transport operations could be performed while the reactor was operating. It was intended that these operations be performed as needed to meet the requirements of the external fuel cycle (i.e., recycle or storage/disposal). The fuel transfer and transport systems will be described later.

The fuel handling system included the gripper and hold-down mechanisms for removing subassemblies from the reactor and installing subassemblies in the reactor; and the transfer arm for transferring subassemblies between the gripper mechanism and storage rack. It also included the rotating plugs and their freeze seals, and certain equipment involved in preparatory operations to permit fuel handling.

The freeze seals for the two rotating plugs provided a combination molten-frozen seal which permitted freezing the upper portion of the seal while retaining the lower region in a molten state. The frozen upper region prevented seal metal loss in the event of a large pressure differential across the seal, while the molten lower region prevented leakage. The entire seal was melted, of course, to permit rotation of the plugs for fuel handling.

After the reactor was shut down, the 12 control rods were released from their individual control rod drive mechanisms and the control rod drive platform was raised 3 inches so that the drives would clear the control rods. The reactor cover hold-down clamps that fastened the cover to the reactor tank were released. The cover elevating columns were raised by two synchronized electric motor-driven lifting mechanisms located on the small rotating plug. In the raised position, the reactor cover engaged pins extending from the cover into the small rotating plug to prevent swinging of the relatively heavy mass (approximately 17 tons) during plug rotation; the reactor cover rotated with the small rotating plug. The cover was raised 9 feet 8 inches to provide clearance below it for removal of subassemblies from the reactor and transfer to the storage rack.

The reactor was now prepared for fuel handling operation. The two rotating plugs were rotated to the proper location to position the gripper over the desired subassembly. Both plugs were supported by roller bearings and rotated by electric motors. In addition to the rotation of the two plugs to the required location, it was also necessary to rotate the gripper unit about its centerline to provide the correct angular orientation of the gripper head.

All operations involved in the fuel handling cycle included provisions for maintaining a known angular orientation of the subassembly. Three locations on the subassembly established its angular orientation:

- The cone-shaped adapter was slotted and engaged a blade in the gripper mechanisms
- The section below the collar was rectangular and engaged a slot at the end of the transfer arm
- The lower nozzles of the subassemblies were slotted and engaged orientation bars in the reactor grid and the storage rack.

Each of these angular orientation controls on the subassemblies was in the same plane. Control of angular orientation and knowledge of angular orientation was maintained at all times during fuel handling.

All of the components and mechanisms involved in these operations were positioned on and in the small rotating plug (Figure 3-24). The small plug, in turn, was located in, and positioned by, the larger plug (Figure 3-25). These various operations were controlled by, and provided feedback for, a variety of circuits requiring a large number of conductors. These were arranged in two multi-conductor cables, one supplying the large plug and one supplying the small plug. To accommodate the rotation of these plugs, these cables were positioned by a festoon cable system, which provided the extension and contraction required as the plugs rotated as shown in Figure 3-26. To minimize the number of conductors involved, the control rod drive conductors were connected only in the reactor operation position of the rotating plugs by multi-conductor electrical plugs that were manually disconnected before fuel handling operations began.

The rotating plugs and gripper head were rotated to the proper position for the particular subassembly to be removed. There was an angular position for each of these three rotating units for each lattice position in the reactor. In preparation for gripping a subassembly, the hold-down mechanism, consisting of a funnel-shaped sleeve, was lowered by an electrically driven screw over the subassembly to be removed. It contacted the six adjacent subassemblies to be removed. This arrangement is shown in Figure 2-9. The hold-down sleeve also acted as a guide for the gripper mechanism.

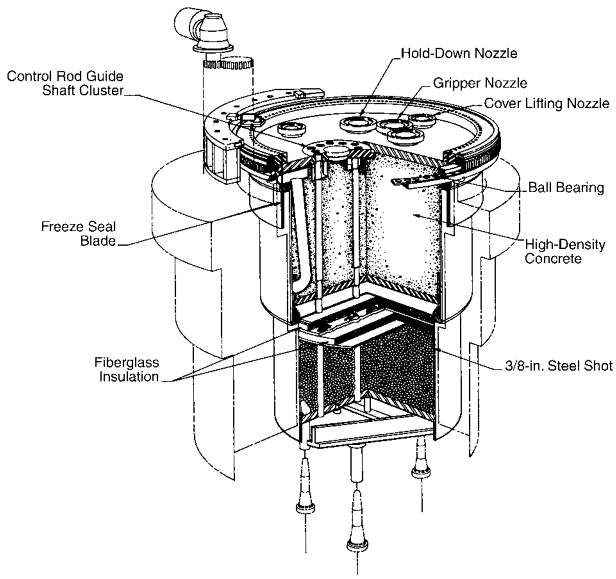


FIGURE 3-24. SMALL ROTATING SHIELD PLUG.

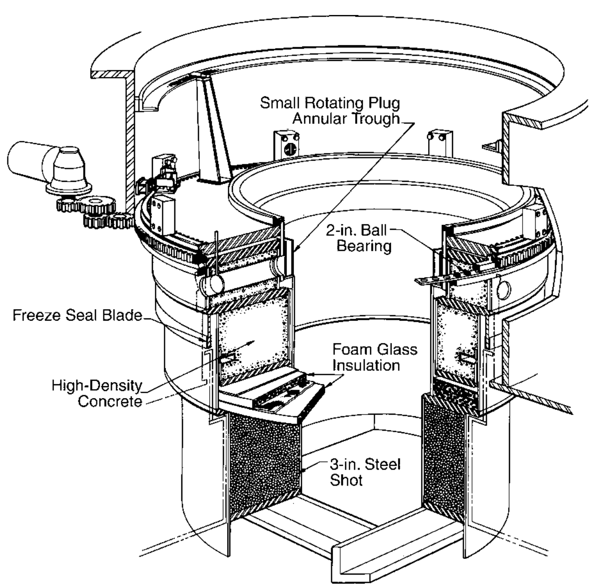


FIGURE 3-25. LARGE ROTATING SHIELD PLUG.

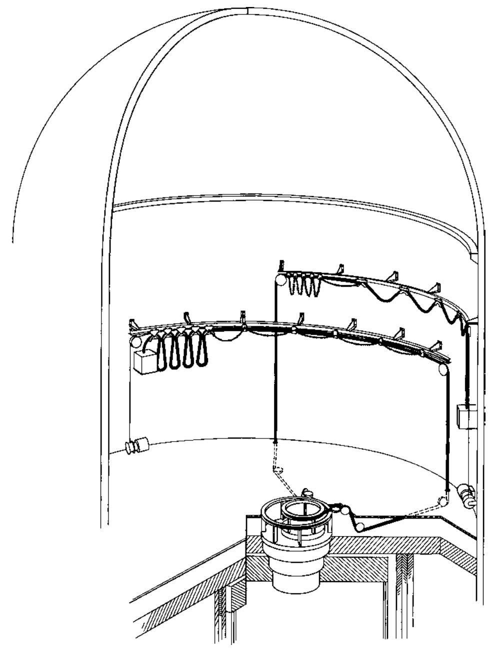


FIGURE 3-26. FESTOON CABLE SYSTEM.

The gripper head was lowered through the hold-down sleeve and contacted the adapter on the subassembly. The gripper device on the lower end of the mechanism gripped the subassembly adapter in the same fashion as the control drive gripper described earlier (Figure 3-20). The orientation blade between the gripper jaws engaged the slot in the conical shaped head. The sensing device also functioned as previously described. The gripping mechanism was moved vertically by an electrically-driven screw drive and the gripper jaws were motor-operated. After the subassembly had been raised out of the reactor, the hold-down tube was raised around the suspended subassembly and provided a lateral support during movement of the two rotating plugs to prevent the subassembly from swinging.

The plugs were rotated to the transfer point, and the gripper head was rotated to the transfer position. The slotted section of the transfer arm engaged the rectangular section of the subassembly adapter to maintain proper angular orientation. The collar of the subassembly adapter fit into the counter-bored recess on the transfer arm holding device when the subassembly was lowered by the gripper mechanism as shown in Figure 3-27. The locking bar on the transfer arm holding device

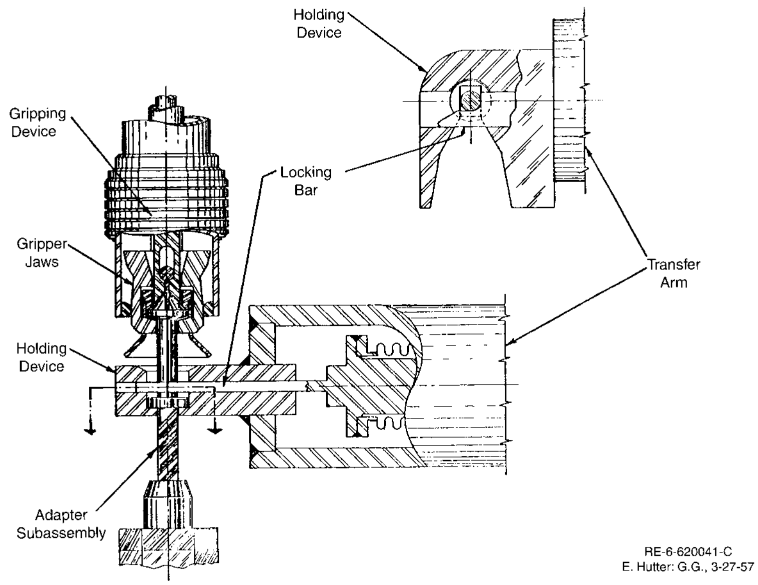


FIGURE 3-27. SUBASSEMBLY TRANSFER.

locked the subassembly positively to the transfer arm. The subassembly was then released from the gripper, the gripper was raised, and the hold-down was lowered below the subassembly.

The transfer arm was rotated through a horizontal arc of about 80 degrees and positioned the subassembly above any one of three concentric rows of storage locations in the storage rack shown in Figure 3-28. The transfer arm was operated manually, and several checkpoints could be felt by the operator. For example, the physical contact between the transfer arm and subassembly at the transfer position was felt by a wiggle test: the transfer arm could not be moved while the subassembly was held by the gripper and hold-down sleeve, and attempting to move it provided a check that the transfer had been made correctly. Similar checks were made between the transfer arm and the storage rack.

The storage rack was a cylindrical vessel providing 75 storage locations for subassemblies in three concentric rows. The storage rack was suspended by a shaft connected to a drive mechanism that provided rotation and vertical movement to the storage rack in the primary tank below the sodium level. An empty storage location was positioned below the subassembly, which was suspended from the transfer arm at the proper angular position. The transfer arm was lowered to provide initial engagement of the subassembly with the storage rack. This was a

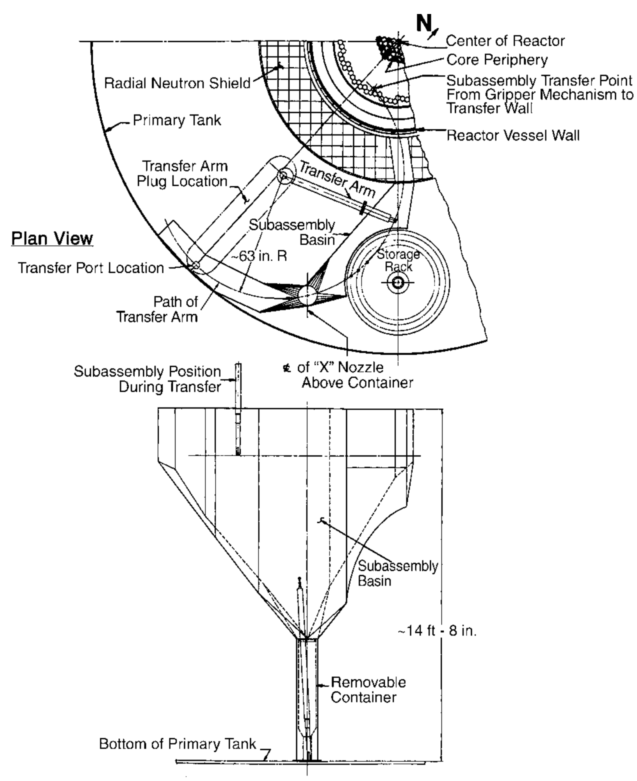


FIGURE 3-28. SUBASSEMBLY BASIN.

manual operation and the operator could feel that the subassembly had entered the storage position. The storage rack was then raised. At the end of the upward movement, the subassembly was lifted from the holding device on the transfer arm.

An additional checkpoint existed here. As long as the subassembly was held jointly by the storage rack and the transfer arm, the transfer arm could not be moved, indicating proper operation of both mechanisms. Following this check, the transfer arm locking bar was released and the transfer arm was rotated to a neutral position while the storage rack was lowered. To remove a subassembly from the storage rack by the transfer arm, the process was reversed.

Although the gripper, transfer arm, and related control circuits were designed to prevent accidental release of a subassembly, provisions were made for recovery. A subassembly catch basin (Figure 3-28), consisting of a funnel-shaped trough, traversed the area under much of the transfer arm path, the arc between the reactor, storage rack, and transfer port. All sides of the funnel sloped toward a depression located directly below an access nozzle (the "X" nozzle) in the primary tank cover. If a subassembly was accidentally released and dropped from the transfer arm at any point along its travel other than over the reactor, the radial neutron shield, or the storage rack, it would drop into the basin and slide to the retrieving position, standing in a near vertical attitude. In this position, the subassembly could be grappled through the access nozzle. Natural convection cooling of this subassembly would occur in a similar fashion as for subassemblies located in the storage rack.

The basin was thoroughly tested after installation by repeated, deliberate dropping of a dummy subassembly from the transfer arm. During the reactor operating lifetime, two subassemblies were dropped out of about 40,000 transfer operations. One subassembly was dropped into the subassembly basin and one on top of the reactor. The first was retrieved as described above; ingenuity and patience retrieved the second. Both experiences will be described as a part of the EBR-II operational experience in other documents.

PRIMARY TANK AND BIOLOGICAL SHIELD

The primary tank, primary tank support structure, biological shield, and shield cooling system comprised an integrated system, designed to meet static load requirements, maintain accuracy of alignment, and contain internal energy release. As shown in Figure 2-12, the tank was surrounded by and supported by the primary structure that included the biological shield.

The primary tank and support structure were separate except at the top. Much of the equipment entering the primary system was large and heavy, requiring adequate support, as did the primary tank itself. The low temperature top structure was designed to support these loads.

The primary structure (Figure 3-29) was also designed to contain the energy release associated with a hypothetical nuclear accident. For design purposes, an energy release equivalent to 300 pounds of TNT at the center of the reactor was assumed. Although the primary tank would be destroyed, the primary structure surrounding the tanks was designed to contain this energy release without failure. It should be noted that these design assumptions were developed in the 1950s and were a very conservative substitution for experience and technology. The operating experience and operational response characteristics of EBR-II would suggest that the EBR-II design was extremely conservative.

The primary tank was constructed with double walls, a tank within a tank to provide maximum reliability of sodium containment. Both the inner and outer tanks were constructed of Type 304 stainless steel. The inner tank had a 26-foot internal diameter. The sidewalls were constructed of 1/2-inch-thick plates, while the tank bottom was constructed of 1-1/2-inch-thick plate. The outer tank sidewalls were constructed of 1/4-inch-thick plates while the tank bottom was constructed of 3/4-inch-thick plate. The 5-inch annulus between the two tanks was filled with an inert gas, which was monitored to detect sodium or air leakage through either tank wall. The outside of the outer tank was insulated to minimize heat loss from the primary system.

The inner tank bottom plate structure was designed to support the reactor tank, the subassemblies, neutron shield, and the entire sodium load. The tank wall transferred this load to

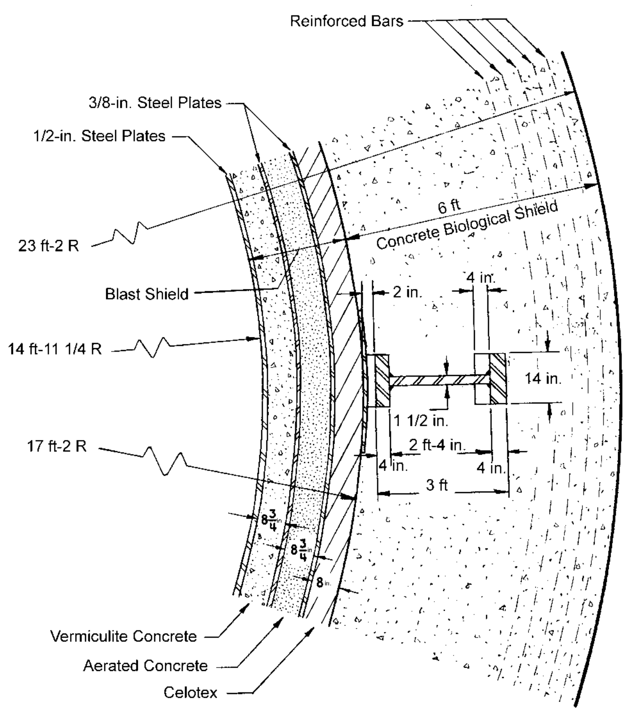


FIGURE 3-29. BLAST SHIELD AND TYPICAL COLUMN DETAIL FOR PRIMARY TANK SUPPORT STRUCTURE.

the top cover where the tank was supported. The outer tank structure was designed to carry only the sodium load in the event of a leak in the inner tank. The bottom of each of the tanks was stiffened with radial beams. The criteria used in the bottom plate structure design were as follows:

1. The inner tank bottom plate structure was designed to support the full load with a maximum deflection of 1/4 inch at a temperature of 750°F. This small deflection was established to minimize misalignment between the reactor and the upper structure of the primary system.

2. The outer tank bottom plate structure was designed to support the uniformly distributed sodium load with an allowable bending stress in the plates and beams of 14,700 pounds per square inch.

The primary tank and its contents, and those components that were connected to the primary tank top cover, were supported by six hangers welded to the top cover beams, which in turn transferred these loads to the top structure beams. Each hanger was supported on a roller to permit radial thermal expansion of the primary tank cover as shown in Figure 2-12.

The primary tank design and the method of support were arranged to provide radial expansion about the vertical centerline of the system. The most critical units, the reactor and the rotating plugs that located the control drives and the fuel handling mechanisms, were located on the physical centerline of the system. Differential vertical expansion was minimized by the use of identical material for all equipment in the system, and maintaining it at the same temperature.

The primary tank support structure (Figure 2-18) consisted of a system of columns and beams that transmitted the loads to the main internal building foundation. In combination with the biological shield, it formed a pressure vessel surrounding the primary tank. The columns were connected to each other by horizontal beams at the bottom, and embedded in the heavily reinforced concrete. These columns were connected at the top to six radial beams which framed into a circular ring (6 inches thick) located on the centerline of the system. With some additional stiffening, this top structure provided the supporting structure for the primary tank and for the major primary system components external to the primary tank. A ring of ordinary concrete (6 feet thick) provided the radial biological shield; the inside diameter was at essentially the same diameter as the inside of the six vertical columns (Figure 3-29).

The radial biological shield and structure was continuous except at an elevation near the top of the primary tank where it was penetrated by several horizontal offset holes, approximately 8 inches in diameter, for the ventilation ducts required for shield cooling. The shield was heated by the heat loss from the primary system, and by energy absorbed in attenuating neutrons and gamma rays. The heat was removed to avoid overheating the steel plates and the concrete.

The shield was cooled by forced circulation of air. It was essentially a recirculation system, however, a fraction of the air was continuously drawn into the system and an equal amount was discharged through the building exhaust system. The shield cooling system operated at a pressure slightly below that of the building atmosphere. This provided in-leakage and also simplified certain areas in the shield that could not be connected to a closed circulation system. The top structure and the shield plugs installed therein were cooled by air drawn from the building atmosphere. The radial shield and the structure below the primary tank were cooled primarily by re-circulated air. Figure 3-30 is a simplified diagram describing the shield cooling system. Air from inside the building was drawn into the primary system through a duct system in the rotating plugs and in the primary top structure, and circulated around the top cover of the primary tank, through ducts in the biological shield into exhaust blowers. It joined air that had circulated through the radial shield and bottom shield air space. The flow then was split into two paths, one to the exhaust stack in the Fuel Cycle Facility, and the other through coolers.

The heat that needed to be removed by the shield cooling system consisted almost entirely of the heat loss from the primary system, the heating in the shield due to neutron and gamma ray attenuation being only a small fraction of the total heat load. The total heat load was approximately 430,000 British thermal unit per hour, of which 415,000 British thermal unit per hour was the heat loss from the primary tank, and approximately 15,000 British thermal unit per hour was due to the neutron and gamma attenuation in the structure and shield.

An air-cooling system of 15,000 cubic feet per minute capacity with a maximum air velocity of approximately 30 feet per second was provided. Reliability of the system was achieved by auxiliary power supplied to the exhaust blowers and coolers. Because of the large heat capacity of the system, brief interruption of the cooling system was not critical.

PRIMARY SODIUM PURIFICATION SYSTEM

A recirculating cold trap system (Figure 3-31) was used for continuous primary sodium purification. This system provided impurity concentrations at or near their greatly reduced solubility limits at temperatures just above the melting point of sodium. Cold trap precipitation was effective in maintaining low concentration of such impurities as sodium hydride, most fission products, and particularly sodium monoxide.

The cold trap consisted of a 500-gallon tank filled with Type 304 stainless steel wire mesh to provide supplementary surface area to enhance sodium crystallization and deposition.

A regenerative heat exchanger was incorporated in the main sodium stream to reduce over-all heat losses in the cold trap system. The cold trap operational temperature of 350°F was maintained by a cold trap coolant loop. Plugging indicators were located on the sodium inlet and outlet sides of the cold trap (plugging/melting temperature increased with impurity concentration of the sodium). They were used to check the efficiency of the cold trapping operations and to provide sampling points for chemical and radiological analysis of the sodium before and after the purification cycle.

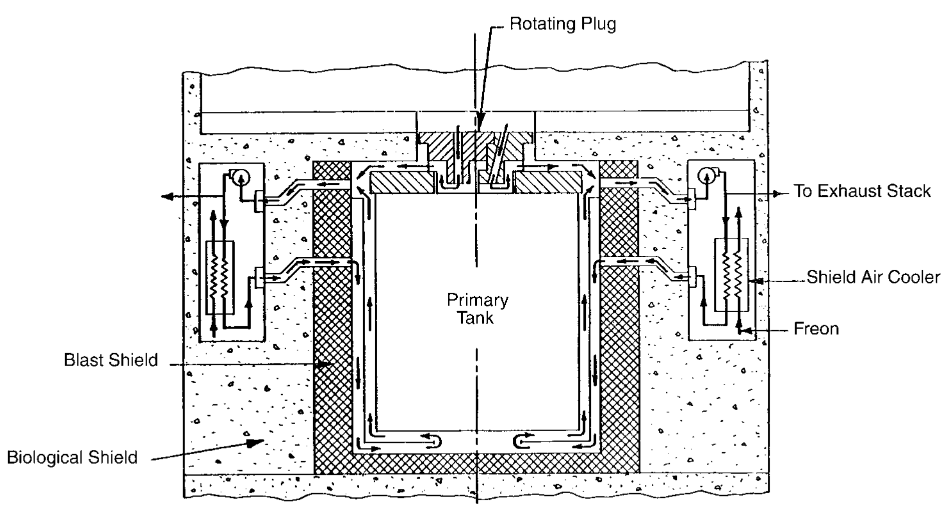


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FIGURE 3-30. SHIELD COOLING AIR SYSTEM SCHEMATIC DIAGRAM.

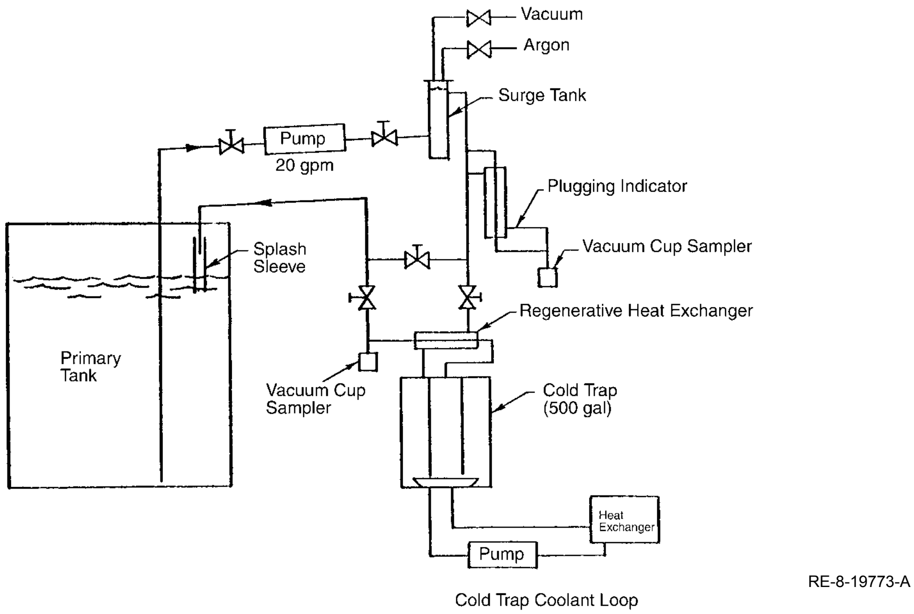

FIGURE 3-31. SODIUM CLEANUP SYSTEM FLOW DIAGRAM.

Parts of the cold trap circuit were below the level of sodium in the primary tank. Since radioactive primary sodium was circulated in the cold trap system, it was essential to eliminate the possibility of an accident or equipment failure resulting in siphoning of primary tank sodium. To avoid this possibility, a surge tank was included in the cold trap inlet line at its highest point of elevation. An argon gas blanket pressure was maintained such that, under static conditions, the sodium level was just below the surge tank discharge opening. With the pump operating, the level rose sufficiently to establish flow. The power supply to the pump was interlocked to a sodium vapor monitor at the cold trap floor level to cut out when a sodium leak was detected, thereby breaking the inlet sodium line at the surge tank. In addition, an argon gas line was provided for positive gas addition to insure breaking the sodium column in an emergency.

INERT GAS SYSTEM

It is necessary to provide an inert gas blanket over sodium. Argon was chosen for this system because of its superiority for pumping, heat transfer, and sealing. To maintain a low level of atmosphere contamination, a gas cleanup system (Figure 3-32) was provided through which the argon could be re-circulated and purified. This system maintained a static argon gas blanket over the primary sodium. The primary tank argon gas blanket pressure is maintained at a positive 1 inch ± ½ inch water pressure differential with respect to building static pressure to prevent excessive loss of blanket gas to the reactor building in the event of a leak. The slight positive pressure also prevents building air from leaking into the blanket gas and contaminating the bulk sodium. The inert gas blanket protects the primary tank sodium from contact with air. Make-up gas was added to the primary circulating gas system, as needed, from the Fuel Cycle Facility argon gas supply system. Excess gas was vented directly through filters to the exhaust stack or to a retention tank for subsequent disposal.

SECONDARY SYSTEM

The secondary system was the non-radioactive sodium heat transfer loop between the radioactive primary system and the steam system (see Figure 3-1). The principal function of this system was to transfer heat from the primary sodium system to the steam system in an efficient manner. The flow rate was 2.5 x 10^6 pounds per hour (approximately 6,000 gallons per minute). The heat exchanger inlet temperature was 588°F and the outlet temperature was 866°F. The principal components of the secondary system, in flow sequence, were the sodium circulating pump, the heat exchanger, the steam superheater and the steam evaporator.

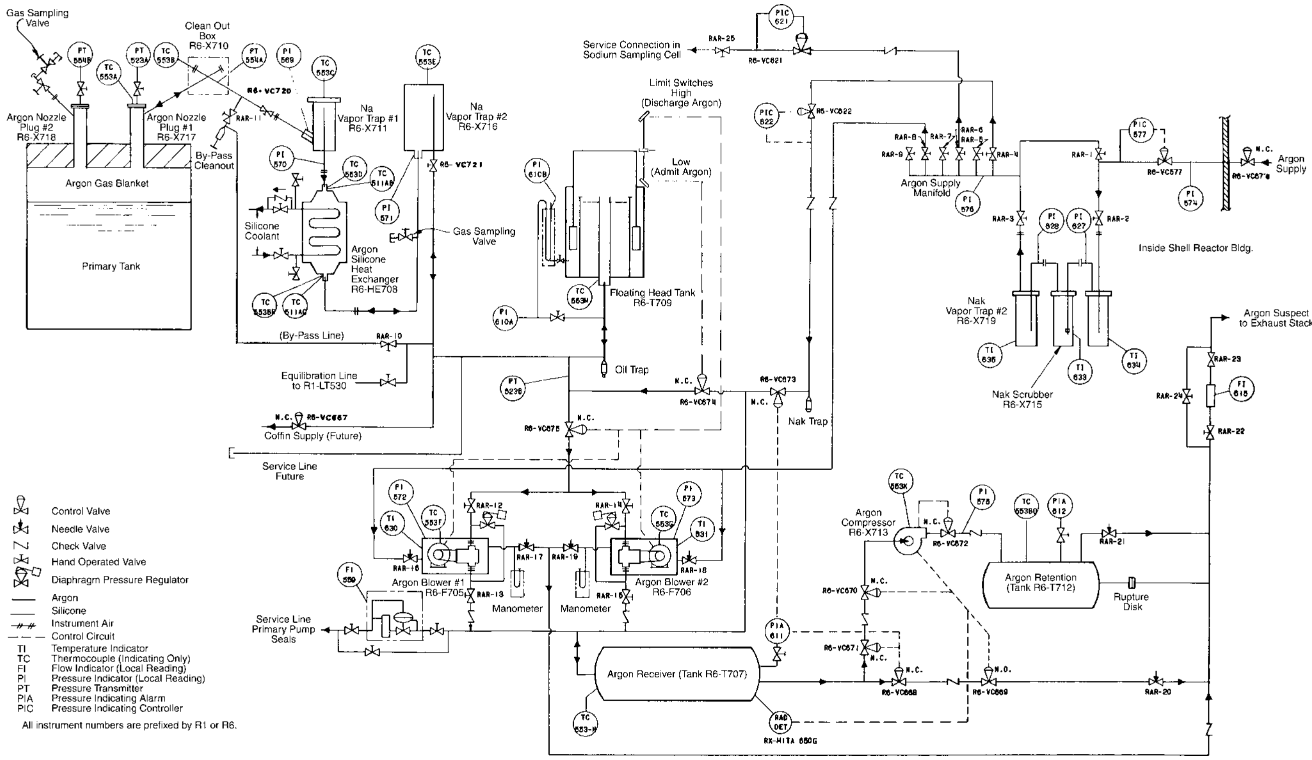

FIGURE 3-32. ARGON BLANKET GAS SYSTEM.

The circulating pump was an alternating current linear induction electromagnetic pump with a capacity of 6,500 gallons per minute at about 53 pounds per square inch. Flow control down to 0 percent of nominal rating, actually to negative reverse flow, was achieved by a generator voltage regulator that used an amplidyne motor-generator set for very accurate voltage control of the main generator output to the pump.

The circulating pump was located in the Sodium-Boiler Plant building which was separated from the Reactor Plant building. This fireproof building also contained the secondary sodium purification system, sodium receiving facilities, and the sodium storage tank. The sodium storage tank was below floor level in this building and the entire secondary system sodium, except that in the heat exchanger, could be drained into this tank.

The surge tank, which was connected into the piping at the circulating pump inlet, maintained a constant head to the pump. The sodium purification system circulated 20 gallons per minute from the storage tank and discharged into the surge tank, ensuring constant level. The overflow returned to the storage tank through an internal overflow pipe in the surge tank. Argon gas at approximately 10 pounds per square inch was provided as an inert gas atmosphere over the sodium in the surge and storage tanks.

The heat exchanger was located within the primary tank in the Reactor Plant. It was suspended from the primary tank cover, and was almost totally submerged in the primary sodium. It was a shell and tube-type exchanger with the secondary sodium on the tube side as shown in Figure 3-12.

The steam generation equipment was located so as to ensure sodium drainage to the storage tank in the sodium plant. The secondary sodium passed through the superheater section and the evaporator section in series (Figure 3-33).

All piping in the secondary system was capable of absorbing thermal expansions due to temperature changes from ambient to 1,000°F. The sodium yard piping was carried on conventional concrete piers fitted with pipe guide or anchor frames as required. The yard piping was heated, insulated, and weatherproofed. Heating was accomplished by 60-cycle induction heating to maintain a temperature above the freezing point of sodium (208°F).

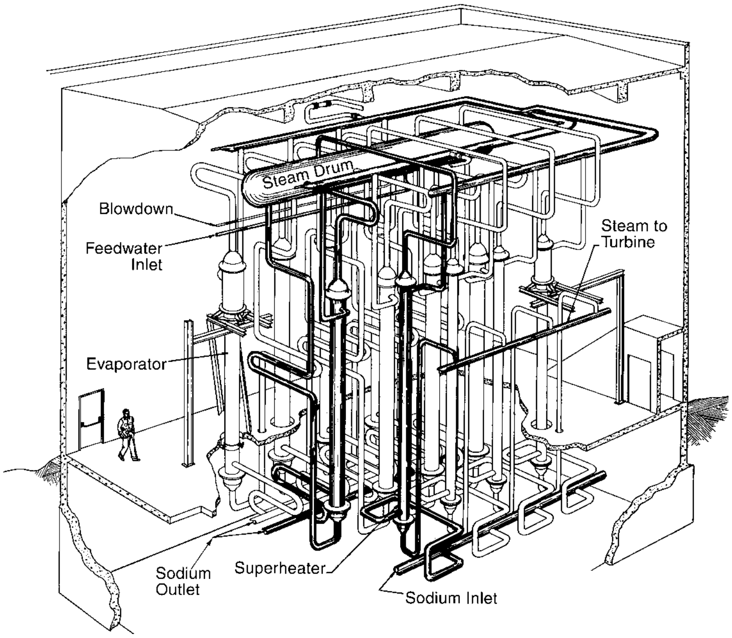

FIGURE 3-33. STEAM GENERATOR.

SODIUM RELIEF SYSTEM

The secondary sodium system included a sodium relief system to accommodate a pressure surge in the event of a sodium-water reaction. This system consisted of two duplex, 10-inch blowout diaphragms. One diaphragm was connected to each 10-inch sodium header that interconnected the superheaters and the evaporators. Each duplex diaphragm, two individual diaphragms in series, was designed to rupture at 100 pounds per square inch. Rupture of the two diaphragms allowed the sodium to flow from the header into a 1,200-gallon pressure relief tank. The tank, in turn, communicated with the atmosphere via two 12-inch lines. Each of these lines was sealed with a rupture diaphragm set for 25 pounds per square inch. The normal sodium pressure in the superheater evaporator headers was about 10 pounds per square inch.

STEAM SYSTEM

The steam system served as a heat sink for power generated in the reactor. Steam was generated at 1,300 pounds per square inch, 850°F from the heat delivered by the secondary sodium system. At 62.5 megawatt thermal reactor output, the steam generator system delivered 248,000 pounds per hour of superheated steam to a conventional 20 megawatt turbine generator system. An induced draft cooling tower provided low-temperature heat rejection.

A steam by-pass system was incorporated around the turbine to permit absorption of all energy produced in the reactor independent of electrical output. The condenser was sized to accept 100 percent of the steam generated.

Steam conditions were selected to provide maximum stability to the heat transfer loops with respect to system temperatures. The saturation temperature of 1,300 pounds per square inch steam (580°F) approximated the minimum temperature of the secondary system. This resulted in a constant high temperature heat sink provided that the steam pressure was maintained constant, which was readily accomplished. The temperature of the secondary sodium seen by the primary sodium coolant system was essentially constant under all conditions of operation.

Achieving reliability of the steam generator unit was a primary objective of EBR-II. High thermal stresses were known to have contributed to failures in other steam generators. In an effort to minimize thermal stresses in the EBR-II steam generator, special feedwater temperature requirements were established. In addition to normal feedwater heating by steam extraction from the turbine, an additional heater supplied with steam directly from the 1,300 pounds per square inch system raised the feed-water temperature further. In this manner, the feed-water was heated to 550°F over the entire load range resulting in a very small temperature difference between the feedwater and the evaporator water (580°F).

The steam generator consisted of a natural circulation evaporating section, a conventional steam drum, and a once-through superheating section. The evaporation section consisted of eight identical shell and tube heat exchangers connected in parallel on the tube side to a horizontal, overhead steam drum with conventional moisture separation internals. Saturated steam flowed from the top of the steam drum downward through vertical shell and tube superheaters, to the turbine generator unit.

The original design of the steam generator system is shown in Figure 3-33 and the original design of the evaporator units and superheater units is shown in Figure 3-34. They were constructed entirely of 2-1/4 percent chromium—1 percent molybdenum material and utilized double-walled tubes. Each duplex tube consisted of two seamless tubes which were individually inspected as single-wall tubes and again inspected as a duplex tube. The units have double-tube sheets at each end; the outer tube was welded to the sodium tube sheet and the inner tube was welded to the steam tube sheet. The space between the two tube sheets communicated directly with the atmosphere. No weld existed in these units with sodium on one side and water and/or steam on the other side. As a result, the only direct path between sodium and water and/or steam was across two seamless tubes that had been individually and jointly non-destructively inspected.

The basic design concept of the evaporators and superheaters was very similar. They were both double tube-double tube sheet designs with sodium and water/steam separated by two barriers. The most significant difference between the two units was the tube diameters and the tube wall thickness. The superheater tubes were smaller in diameter and had a thinner wall. This difference caused a welding problem that impacted the construction of the units.

Difficulty was encountered in fabrication of the superheater units; more specifically, the outer tube-to-sodium tube sheet welds (see inset, Figure 3-35). Many sound and reliable welds were made, but not consistently. This welding problem did not prevail during fabrication of the larger tube evaporators. The smaller tube diameter and thinner wall of the super heater tube could not be made reliably with the welding techniques available at the time. An alternate method of superheating was selected.

Spare parts of evaporator units were available for the fabrication of two additional evaporators. These two units were modified to serve as superheaters. The major difference was the addition of a core tube in each evaporator tube with a 0.812-inch outside diameter that provided increased steam velocity in the 0.125-inch steam annulus (see Figure 3-35).

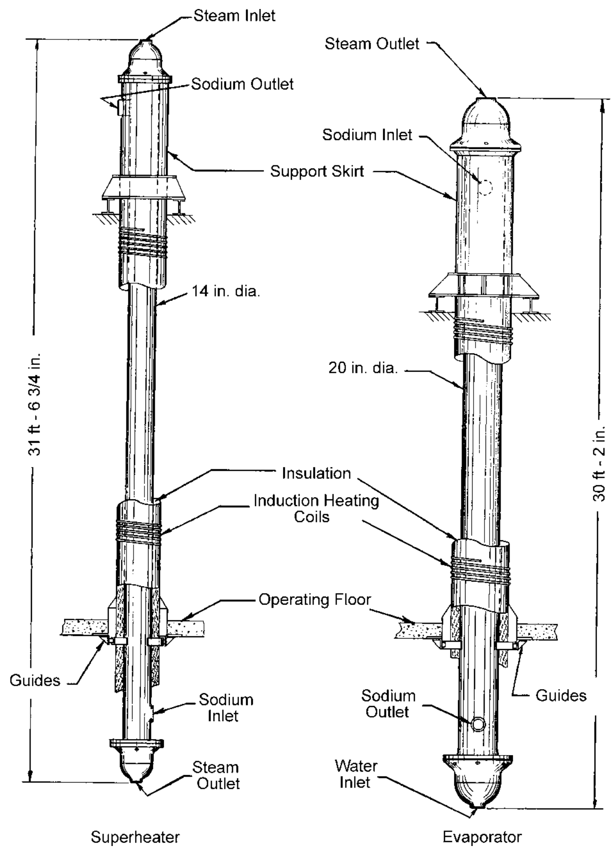

FIGURE 3-34. SUPERHEATER AND EVAPORATOR ASSEMBLIES.

The use of two modified evaporators as substitute superheaters resulted in a reduced steam temperature and a slight increase in the moisture content in the final stage of the turbine. The slight moisture increase did not seriously affect operation of the machine during its 30-year operating lifetime.

At the time this decision was made, it was considered a temporary solution to place the plant in operation, but satisfactory operation of the plant was achieved without any change and the temporary fix was made permanent. As noted earlier, achievement of high thermal efficiency was not an EBR-II primary objective, but reliable operations was.

Owing to the external similarity between the evaporators and superheaters, only minor modifications to the building, supporting structure, and piping were required to effect use of the modified evaporators.

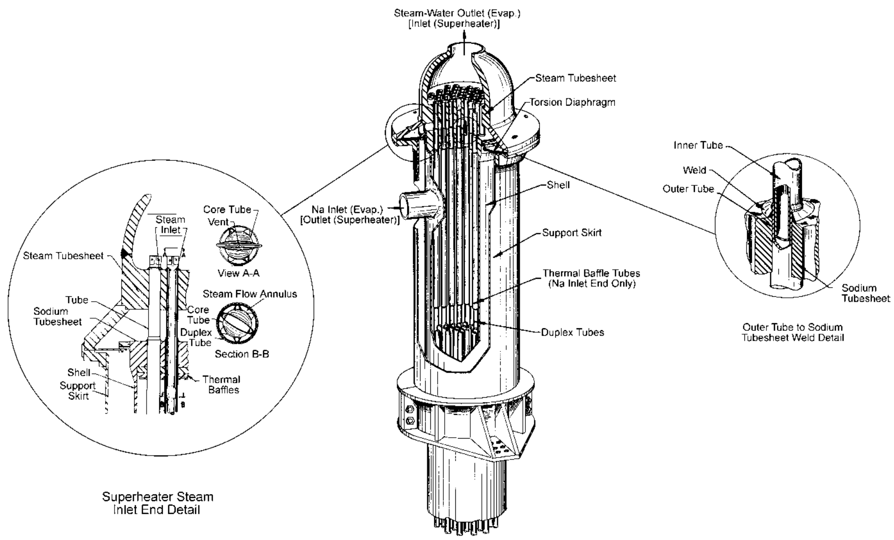

FIGURE 3-35. EVAPORATOR AND "MODIFIED SUPERHEATER" DETAILS.

FUEL TRANSFER AND TRANSPORT SYSTEMS

As described earlier, the fuel handling system delivered subassemblies to the storage rack in the primary tank where they were stored and ready for recycle or other disposition. Subsequent operations involved a two-step process; fuel transfer and fuel transport. These operations could be performed with the reactor in operation. They were independent of reactor operation and were coordinated with the fuel cycle. Although very similar procedures and processes were involved in the transfer and transport of fuel assemblies for either recycle or other disposition, only the operations involved with recycle will be described here. The equipment and components involved in this process are depicted in Figure 3-36. Restricted operation is indicated since the reactor is shown in the operating configuration.

FUEL TRANSFER

The fuel transfer system moved subassemblies between the storage rack in the primary tank and the inter-building coffin. The mechanical components included the storage rack, transfer arm, transfer port, fuel unloading machine and the inter-building coffin (Figure 3-36).

The transfer port provided access to the inside of the primary tank through the transfer arm nozzle. It provided the link between the transfer arm inside the primary tank and the fuel unloading machine, which operated on the main floor above the transfer port. The transfer port was basically a large, manually operated valve, normally closed, which was opened while attached to the fuel unloading machine to permit subassembly transfer. It had provisions for argon gas purging as required for fuel transfer operations.

The fuel unloading machine (Figure 3-37) was an electro-mechanical device that transferred fuel subassemblies from the transfer arm inside the primary tank to the inter-building coffin outside the primary tank. The machine was basically a shielded container-carriage assembly, mounted on a set of tracks on which it traveled between the transfer port and the inter-building coffin. The internal mechanisms included a gripping device, and an argon gas circulating system.

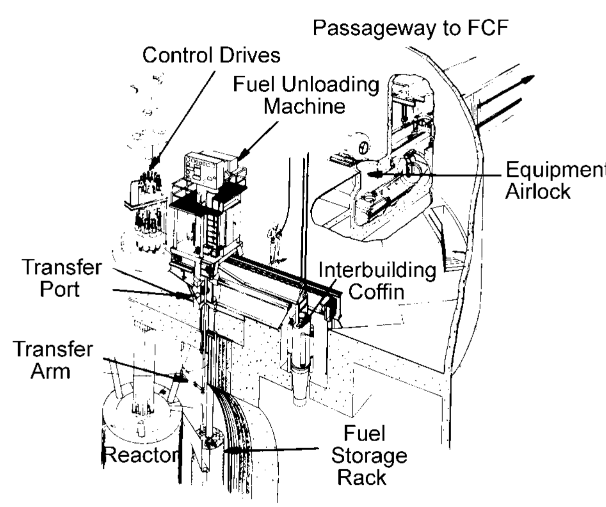

FIGURE 3-36. FUEL HANDLING SYSTEM.

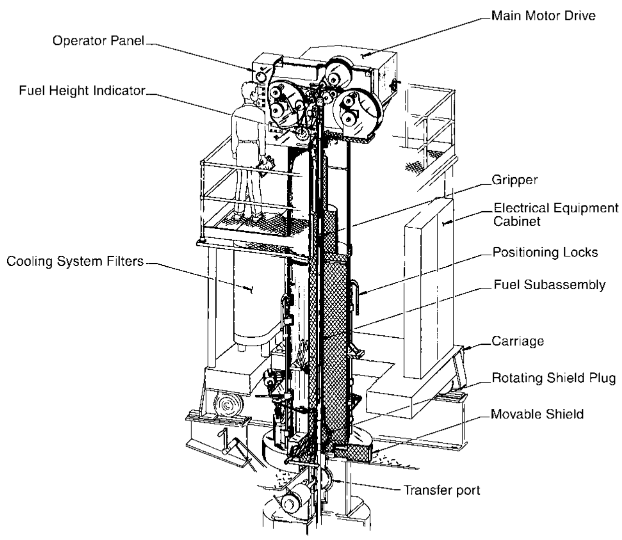

FIGURE 3-37. FUEL UNLOADING MACHINE.

Briefly, the sequence of fuel subassembly transfer was as follows. An irradiated subassembly was removed from the storage rack by the transfer arm and aligned directly under the transfer port. The fuel unloading machine was positioned over and sealed to the transfer port. The transfer port was purged to provide a total argon gas environment, i.e., in the primary tank, the transfer port and the fuel unloading machine. The gripping device was lowered through the transfer port to the level of the transfer arm to engage the subassembly. The transfer arm was disengaged from the subassembly. The subassembly was lifted into the fuel unloading machine, transported to and lowered into the inter-building coffin. The reverse procedure was employed to transfer a recycled subassembly into the storage rack from the inter-building coffin.

The argon gas circulation system on the fuel unloading machine was used to:

- Drain excess sodium from the subassembly as it was removed from the primary tank
- Cool the spent fuel during the transfer to the inter-building coffin
- Preheat a recycled subassembly before insertion into the primary tank sodium.

The transition from sodium cooling to inert gas cooling occurred during the fuel transfer process as the subassembly was raised into the fuel unloading machine. The reverse process occurred when a reprocessed subassembly was preheated and transferred from the fuel unloading machine to the storage basket.

INTER-BUILDING FUEL TRANSPORT SYSTEM

Inter-building fuel transport involved the transportation of a fuel, blanket, or other, subassembly from the Reactor Plant to the Fuel Cycle Facility or the transportation from the Fuel Cycle Facility to the Reactor Plant. The primary vehicle for performing this task was the inter-building coffin. Transporting this 15-ton carrier required cranes, carriages and controlled passage through the equipment air lock between the two buildings.

The inter-building coffin was a portable, sealed, shielded vessel with an integral argon gas (or air) cooling system. The cooling units on the inter-building coffin were battery-powered to ensure continuous operation in the event of transport difficulties or power failure during transit inside and between the buildings. The primary functions of the inter-building coffin were to provide radiation shielding of subassemblies during transport and to provide cooling to remove the heat generated by fission product decay.

A simplified routing of the inter-building coffin is shown in Figure 3-38.

The first step in the fuel cycle after the inter-building coffin arrived in the Fuel Cycle Facility involved removal of the sodium adhering to the subassembly. This was radioactive primary sodium. The predominant isotope was sodium-24, which has a short half-life of 15 hours. The subassembly was blanketed by argon gas that was circulated to cool the spent fuel, and was the environment in which the process began. The sodium was removed from the subassembly in the inter-building coffin in the following steps:

1. Admit oxygen diluted by nitrogen to the circulating gas system to oxidize the sodium adhering to the subassembly.
2. Admit water vapor in nitrogen to allow further reaction with the sodium.
3. Flow water through the coffin and subassembly to remove the oxidized sodium and assure that all sodium was oxidized and removed.

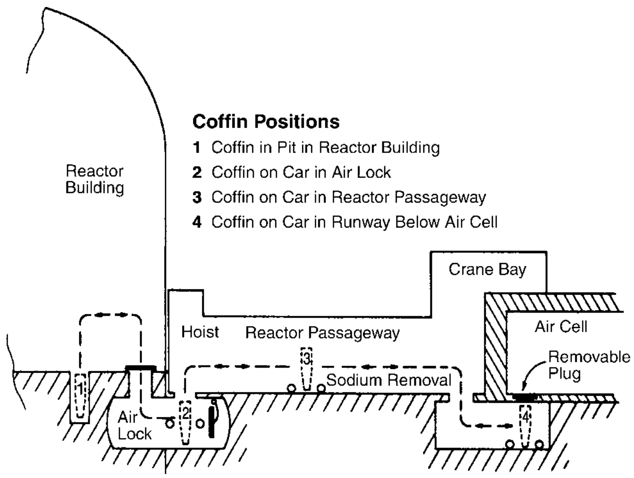

FIGURE 3-38. MOVEMENTS OF SUBASSEMBLY COFFIN BETWEEN REACTOR AND FUEL CYCLE FACILITY.

4. Immediately after the water wash, blow out the water and dry the inter-building coffin and subassembly with a stream of air.

At this point the subassembly was in an air environment and air became the circulated coolant to remove fission product decay heat. The gas cooling system on the inter-building coffin continued to perform this function. Argon gas or air were the cooling medium as appropriate.

The final step in the transport process involved delivery of the subassembly to the air cell for dismantling. The inter-building coffin was lowered onto a cart below and adjacent to the air cell and then moved into position below the air cell, permitting the air cell crane to lift the subassembly into the air cell. During this transfer process, the gaseous medium in the inter-building coffin continued to be air that was circulated through the subassembly to remove fission product decay heat. This cooling process was continued until the subassembly was dismantled and the individual fuel elements were separated from the close packed hexagonal array that existed in the subassembly. When separated, the fuel elements cooled in ambient air and forced circulation cooling was no longer necessary.

FUEL RECYCLE SYSTEM

Fuel recycle was accomplished in the Fuel Cycle Facility which was designed for reprocessing the fuel material discharged from EBR-II by pyrometallurgical methods. The fuel alloy was enriched uranium-5 percent fissium and contains about 46 weight percent uranium-235. The major processes involved are summarized here. A detailed description of the processes and the equipment involved are described in “The EBR-II Fuel Cycle Story,” 1987, by Charles E. Stevenson.

The Fuel Cycle Facility included an argon-atmosphere cell where fuel reprocessing and fabrication could be performed in an inert gas environment, an adjacent air-atmosphere cell where fuel subassemblies could be assembled and disassembled, and an operating area for personnel that surrounded the two cells. Because of the high levels of radioactivity involved, the fuel handling and processing had to be accomplished by remote operation of processing and supporting equipment.

Remote processing was accomplished with the aid of bridge cranes, electromagnetic bridge manipulators, and master-slave manipulators. Transfer ports and air locks were provided for the transfer of materials and equipment into the argon cell and between the two cells. The walls between the cells and the operating areas were heavily shielded, and viewing was provided through thick shielding windows.

EBR-II initial operations included disassembly of fuel subassemblies and their constituent fuel elements, fuel purification, refabrication of fuel elements, and reassembly of the subassemblies for reloading into the reactor. Specific operations included: subassembly disassembly, fuel element decanning, chopping of fuel pins, melt refining, oxidation of skull material retained in melt refining crucibles, injection casting of fuel pins, final pin fabrication, canning of fuel pins, sodium bonding and bond testing, fuel element inspection and testing, and assembly into fuel subassemblies. These operations required a large number of supporting activities which also have been described in Stevenson (1987), including the processes and equipment involved. These include:

- Fuel movement and storage
- Sampling and analysis of fuel and waste
- Preparation, handling, and storage of liquid, solid, and gaseous radioactive wastes
- Disposal of scrap and unrecoverable fuel

- Special controls to avoid criticality and to provide material accountability.

The initial EBR-II fuel cycle consisted of the recycle of enriched uranium containing five weight percent fission products. This alloy was named fissium (Fs) and contained the noble metal fission products that were not removed by the EBR-II pyrometallurgical fuel recycle process. The initial alloy composition was established to approximate the expected equilibrium composition of the alloy and thereby reduce the changes in composition and properties that would occur as the fuel was repeatedly recycled. The EBR-II was the first power reactor system in the United States power demonstration program to operate on a closed fuel cycle utilizing recycled fuel. Many new developments, both in procedures and equipment, were required to perform the various steps in the fuel cycle. Actual plant experience with remote operations and equipment was needed to demonstrate feasibility.

The Fuel Cycle Facility through-put was based on reactor operation at 62.5 megawatt thermal and average burn up of 2 atom percent. This operating mode would produce about 3,130 grams per day of spent fuel. A typical melt refining charge consisted of 10 to 12 kilograms. This was a batch process, but there was adequate storage capability in the primary tank storage rack, and batch size was not critical.

The Fuel Cycle Facility incorporated a unique design developed specifically to perform the processes and operations involved in the EBR-II fuel cycle as summarized in Figure 3-39. This flow diagram depicts all of the direct and supporting activities which had been identified at the time. It defined a total recycle program which attempted to describe a development program for total fast breeder reactor fuel and blanket recycle. Much of the supporting development, such as the blanket material process were not developed, but significant advances have been made in the development of similar fuel recycle of plutonium-uranium metal fuel alloys. Also, it would appear that recycle of these alloys could be performed in an EBR-II-type Fuel Cycle Facility (as originally conceived in the EBR-II plant concept). Development of applicable processes and equipment has continued intended for operation and demonstration in the Fuel Cycle Facility which has been upgraded for this purpose.

The starting point for the EBR-II fuel cycle was the delivery of a spent fuel assembly, which has had the residual sodium removed as a part of the fuel transport, to the Fuel Cycle Facility air cell. As a part of this delivery process, forced convection cooling by air circulation through the subassembly had to be maintained. This circulation was provided in the inter-building coffin and in the air cell dismantler where the first operation was performed. The subassembly hex can was cut at the lower adapter and pulled off exposing the cluster of fuel elements. The fuel elements were removed from the subassembly by rows and placed flat in trays, approximately 30 per tray. In this configuration no forced cooling was required. After appropriate inspection the fuel elements were transferred on the trays through an air lock to the argon cell, they continued to cool naturally in the argon gas environment.

EBR-II fuel processing began by separating the fuel pin from the fuel tube. The fuel tube was sheared at each end at points which also released the spiral spacer wire which was welded to the tube at each end of the wire. The tube was then peeled from the pin, which was accomplished by spiral cutting the tube into a narrow continuous strip. As the clad was removed, the exposed pin was chopped into approximately 1-1/2 inch lengths to provide suitable feed for the next process which involved melt refining.

The chopped pins were fed into a zirconia crucible used for the melt refining operation. Approximately 10 to 12 kilograms of fuel constituted a normal charge. The crucible was heated to 1,400°C and the charge held in the molten state at this temperature for about three hours. During that period, gases and some fission products which volatilize, were removed and some fission products were oxidized. At the end of the heating period, the melt was poured into a graphite crucible and formed a metal ingot. About 90 to 95 percent of the fuel plus noble metals were in the metal ingot, and 5 to 10 percent of the charge remained in the skull and was recovered in a separate process. More detailed descriptions are given in Stevenson (1987). It should be noted, however, that the total recovery of fuel alloy from both processes (i.e., the ingot plus processed skull was 99.8 to 99.9 percent.

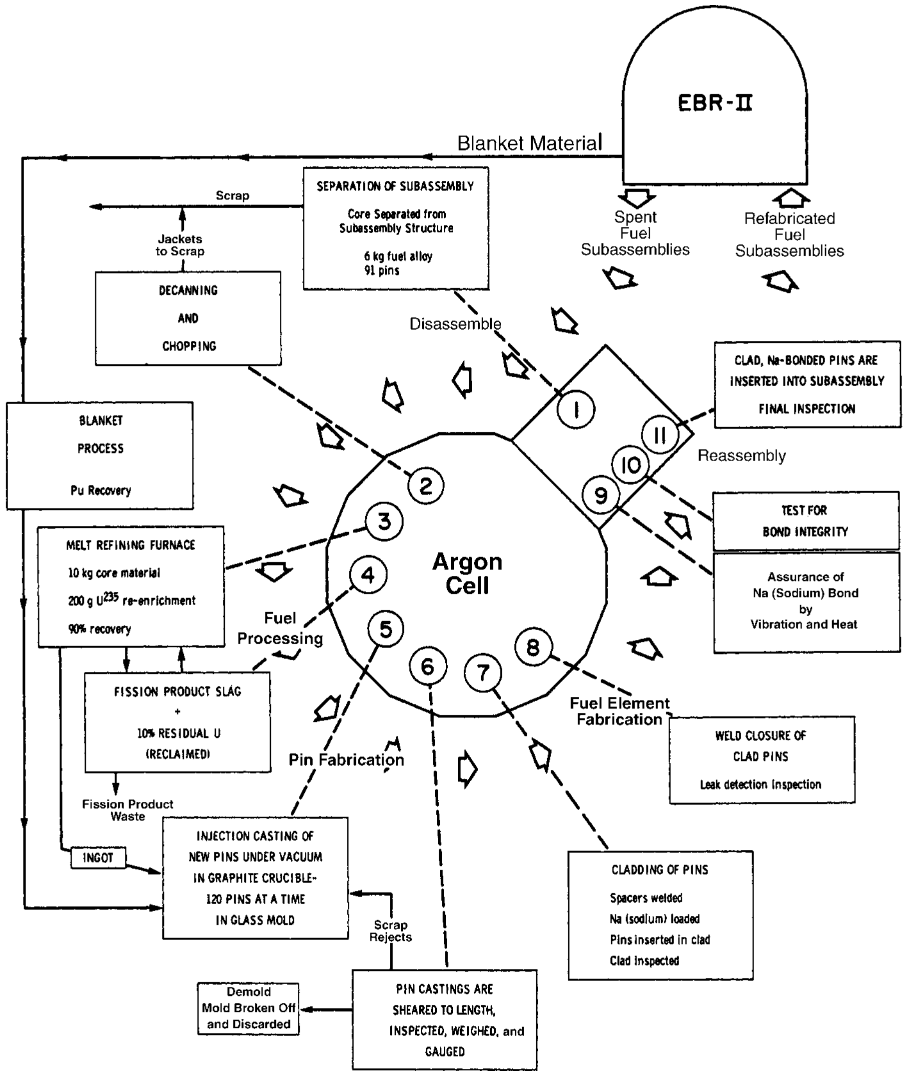

FIGURE 3-39. EBR-II FUEL CYCLE FLOW.

The recovered material was used as feed for the injection casting process. This was a unique machine which produced precision castings in one operation. The casting furnace employed a high-frequency induction heating system and a graphite crucible. The fuel alloy was cast directly into precision Vycor glass molds by a vacuum/gas pressure system. The inside diameter of the molds and the graphite crucible were coated with thoria (ThO_2). The injection casting process involved a series of carefully controlled operations. The furnace assembly, including the crucible containing the fuel alloy and the Vycor glass molds were contained in a gas tight enclosure which was evacuated. The fuel alloy in the crucible, located directly below the cluster of molds, was heated to approximately 1,350°C. The crucible was raised by a pneumatic cylinder immersing the open end of the molds in the molten fuel alloy. The furnace was then pressurized with inert gas and after the alloy solidified in the molds (about two seconds), the crucible was lowered. All of these operations required very careful and accurate control. A normal run involved a fuel alloy charge of 11 to 14 kilograms and approximately 100 Vycor glass molds. The total process was accomplished in about eight hours. The castings were finished pins, except that they required cutting to correct length which was accomplished by shearing. The sheared ends and the heel remaining in the graphite mold were used as feed material in subsequent runs. Approximately 44,000 pins were cast for the initial operating phase of EBR-II.

The finished pins, after detailed measurement, inspection and recording of data were assembled into fuel elements. This step was preceded by the preparation of the fuel element tube assembly which included the tube, the attached lower adapter (hook) and the spiral spacer wire. This assembly was delivered to an argon-atmosphere glove box adjacent to and attached to the argon cell via an air lock penetration. In the glove box, the sodium to provide the bond between the fuel pin and the fuel element tube was installed. This was done by preparing a sodium extrusion slightly smaller in diameter than the inside diameter of the fuel element tube and of the proper length, thus providing the proper volume of sodium.

These fuel tube assemblies were then transferred to the argon cell where the fuel pin was installed, the sodium was melted to permit the fuel pin to settle to the bottom of the tube assembly and the top plug was installed and welded. The top plug also served as a restrainer to prevent the fuel pin from protruding above the sodium bond level. The weld closure was also a very unique concept. It consisted of a flanged plug the same outside diameter as the fuel tube. It also included a small projection in the center for welding as shown in Figure 3-40. The weld was accomplished by a condenser discharge through a tungsten electrode positioned directly above the projection in the center of the plug. The entire top end was fused as shown. The concept and process were developed at Argonne.

FIGURE 3-40. AN ASSEMBLED FUEL ELEMENT BEFORE AND AFTER WELDING.

The completed sealed fuel elements were then transferred through an air lock to the air cell where a series of processes and tests were preformed to ensure the completeness and accuracy of the sodium bond. This was necessary to ensure the reliability of heat transfer from the fuel to the primary sodium coolant. The completed fuel elements were also leak tested to ensure that the closure was leak tight.

The completed fuel elements were then delivered to the subassembly station for the final assembly operations. Here again the maximum permissible preassembly operations, not involving significant radiation, were employed to produce two major preassemblies; the lower preassembly consisted of the lower adapter, lower blanket section, and the grid to which the fuel elements would be attached; and the upper preassembly consisted of the upper adapter, upper blanket section, and the hexagonal subassembly tube. These two preassemblies are shown in Figure 3-41. The lower preassembly was placed into the assembly machine shown in Figure 3-42. The fuel elements were slid onto the parallel T-strips that constituted the lower fuel element grid. This grid consisted of 11 parallel T-strips to which the fuel elements had to be attached in the proper sequence. Sequencing and proper angular orientation of the fuel element were controlled by the element loading block (Figure 3-43) which permitted the process to begin with a much wider spacing than the grid strips. Each fuel element was first placed into the proper, controlled, position in the loading block and then slid on the guide wire to the grid strip. This action was performed by the use of two master slave manipulators augmented by auxiliary devices to assist specific actions. These devices were incorporated into the fuel element assembly machine (Figure 3-42). The operations were visible through 5-foot-thick shielding windows. Note that provisions were made for cooling the fuel elements during the assembly process.

After all 91 fuel elements were installed and properly positioned and supported, the upper preassembly was lowered over the fuel section down to the lower adapter. At this position the hex tube was spot welded to the lower adapter. The completed subassembly was checked for dimensional correctness and straightness and delivered to the inter-building coffin. It was then transported to the Reactor Plant and transferred to the storage rack in the primary tank. This process was essentially the reverse of the delivery

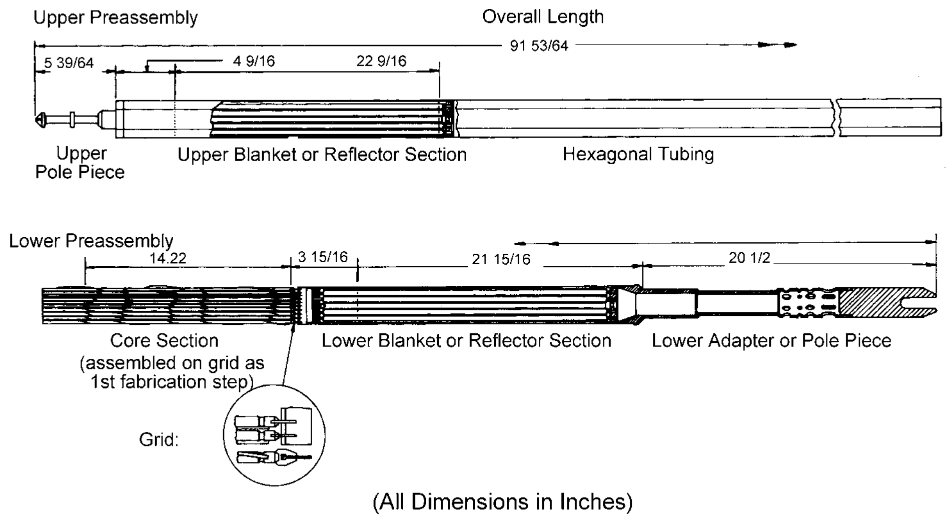

FIGURE 3-41. THE TWO PREASSEMBLED COMPONENTS OF AN EBR-II CORE SUBASSEMBLY.

of an irradiated subassembly to the Fuel Cycle Facility, except that there was no need to wash the subassembly to remove sodium, but there was a need to preheat the subassembly before immersing it into the 700°F primary sodium.

The EBR-II Fuel Recycle System demonstrated that fuel recycle for a power reactor system need not produce a pure, clean product as were required for most military products. It also demonstrated that a relatively small facility, as compared to other purification processes, could accommodate the needs of a power reactor. The Fuel Cycle Facility probably had the capability of processing 5 to 10 times the output of EBR-II. It is quite probable that a facility utilizing this type of process/fabrication cycle could serve more than one reactor. This becomes an exciting possibility when considering the nuclear power park concept.

LABORATORY AND SUPPORT FACILITIES

The Laboratory and Service Building was the facility in which analytical and control support activities were conducted. Six analytical caves (hot cells) were provided to handle small radioactive samples, including irradiated fuel samples and radioactive samples of the fuel elements and process materials. These samples included fuel alloy, cladding, oxidized skull, and scrap. Analytical facilities were also provided to support operation and control of argon and sodium systems, and a variety of waste processes. These facilities and operations are described in Stevenson (1987).

The Laboratory and Service Building also provided personnel support facilities, library, cafeteria, graphic arts, and offices. These proved to be quite inadequate, and during the operating life of the plant required almost constant expansion. This was due, at least in part, to the broadened and enlarged mission of EBR-II.

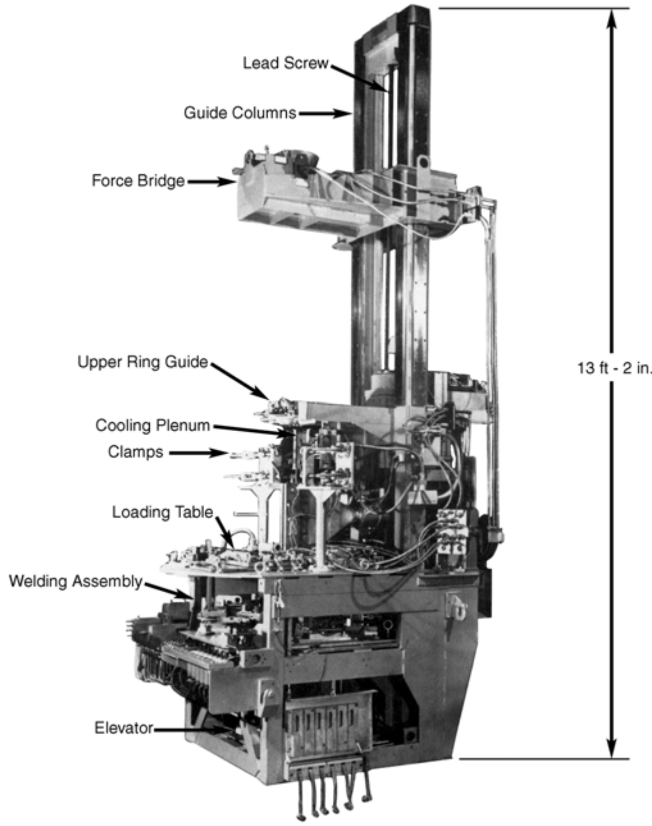

FIGURE 3-42. UNIVERSAL FUEL ELEMENT ASSEMBLY MACHINE.

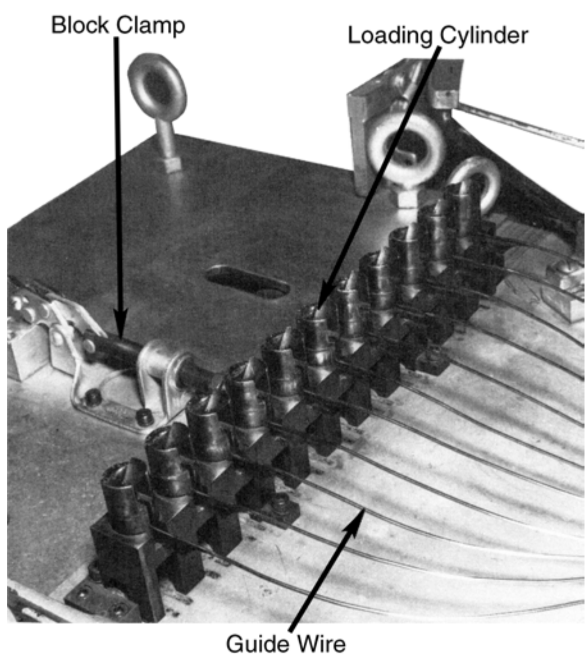

FIGURE 3-43. FUEL ELEMENT LOADING BLOCK.

APPENDIX A

ORIGINAL EBR-II PERFORMANCE DATA AND STATISTICS AND CHRONOLOGY OF PLANT HISTORY

ORIGINAL EBR-II PERFORMANCE DATA AND STATISTICS

General	Design
Heat output (MW)	62.5
Gross electrical output (MW)	20
Primary sodium temperature, to reactor (°F)	700
Primary sodium temperature, from reactor (°F)	883
Primary sodium flow rate, through reactor (gpm)	9,000
Primary sodium maximum velocity, in core (ft/sec)	23.8
Primary system sodium capacity (gal)	89,000
Secondary sodium temperature, to heat exchanger (°F)	588
Secondary sodium temperature, from heat exchanger (°F)	866
Secondary sodium flow rate (gpm)	5,890
Steam Generator	
Output (lb/hr)	250,000
Steam temperature (°F)	837
Steam pressure (psig)	1,300
Feedwater temperature (°F)	550
Turbine Throttle Conditions	
Steam flow (lb/hr)	195,300
Steam temperature (°F)	837
Steam pressure (psig)	1,250
Reactor Data	
Subassemblies	67 Subassembly Core
Core	53
Control (rod and thimble)	12
Safety (rod and thimble)	2
Inner blanket	60
Outer blanket	510
Total	637
Configuration	Hexagonal
Dimension across flats (in.)	2.295
Hexagonal tube thickness (in.)	0.040
Structural material	304
Lattice spacing (pitch) (in.)	2.320
Fuel Elements (Pin-Type, Sodium Bonded)	
Fuel pin diameter (in.)	0.144
Fuel pin length (in.)	14.22
Fuel tube, outside diameter (in.)	0.174
Fuel tube wall thickness (in.)	0.009
Thickness sodium bond annulus (in.)	0.006
Elements per subassembly	91

ORIGINAL EBR-II PERFORMANCE DATA AND STATISTICS (CONTINUED)

Upper and Lower Blanket Elements (Pin-Type, Sodium Bonded)	
Blanket pin diameter (in.)	0.3165
Blanket pin length (total) (in.)	18.0
Blanket tube, outside diameter (in.)	0.376
Blanket tube wall thickness (in.)	0.022
Thickness sodium bond annulus (in.)	0.008
Blanket elements per subassembly (each end)	18
Control and Safety Rods	
Configuration	Hexagonal
Dimensions across flats (in.)	1.908
Fuel elements	Same as Core S/A
Fuel elements per rod	61
Radial or Outer Blanket Elements (Pin-Type, Sodium Bonded)	
Blanket pin diameter (in.)	0.433
Blanket pin length (total) (in.)	55.0
Blanket tube, outside diameter (in.)	0.493
Blanket tube wall thickness (in.)	0.018
Thickness sodium bond annulus (in.)	0.012
Blanket elements per subassembly	19
Fuel (Enriched Uranium)	
Total core loading (kg)	385
Uranium-235 enrichment (a/o)	48.4
Critical mass uranium-235 (clean, full power) (kg)	172
Total mass of uranium-235 in core (kg)	176
Fuel Alloy Composition (Fissium)	U-5Fs
Uranium (wt%)	95.0
Zirconium (wt%)	0.10
Molybdenum (wt%)	2.44
Ruthenium (wt%)	1.94
Rhodium (wt%)	0.28
Palladium (wt%)	0.19
Niobium (wt%)	0.01
Silicon (wt%)	0.04

ORIGINAL EBR-II PERFORMANCE DATA AND STATISTICS (CONTINUED)

Nuclear Data	
Total fissions per cc/sec at center of core	3.7×10^{13}
Uranium-235 fission rate at center of core (fission/g-s)	8.93×10^{12} (at 50 MW)
Neutron Energy Distribution at Center of Core	
Flux above 1.35 MeV (n/cm^2-sec)	0.69×10^{15}
Flux below 1.35 MeV (n/cm^2-sec)	2.86×10^{15}
Total neutron flux (n/cm^2-sec)	3.55×10^{15}
Prompt neutron lifetime (sec)	8×10^{-8}
Reactor Control	
Full-Flow Power Coefficients	
0–62/5 MW (Δk/k)/MW	-3.5×10^{-5}
0–25 MW (with bowing), (Δk/k)/MW	-3.5×10^{-5}
25–34 MW (with bowing), (Δk/k)/MW	$+1.0 \times 10^{-5}$
34–62.5 MW (with bowing), (Δk/k)/MW	-4.0×10^{-5}
Doppler effect—average (Δk/k)/°C	$<+0.4 \times 10^{-6}$
Isothermal temperature coefficient, (Δk/k)/°C	-3.6×10^{-5}
Total Reactivity Worth	
Control rods ($)	7.06
Safety rods ($)	1.91
Control Rods	
Total	12
Operating drive (each rod)	Rack and pinion
Velocity (in./min)	5
Total movement (in.)	14.0
Scram drive	Pneumatic
Safety Rod	
Total	2
Operating drive	Rack and pinion
Velocity (in./min)	2.0
Total movement (in.)	14.0
Scram drive	Gravity

KEY EVENTS IN EBR-II LIFETIME

December 3, 1953	*Preliminary Proposal and Feasibility Report* (ANL-LJK-10), "PBR Development Program" completed. The feasibility of an integrated fast breeder reactor power plant and fuel processing facility was determined, and a preliminary design was proposed.
March 1954	Original Construction Project Data Sheet issued.
July 11, 1955	Original authorization of funds (Public Law 84-141).
November 15, 1956	Title I Engineering started.
April 30, 1957	Title I Engineering completed.
May 15, 1957	*Hazard Summary Report* (ANL-5719) completed.
May 1957	Title II Engineering, H. K. Ferguson Co., Architect Engineer, started.
August 21, 1957	Authorization of funds revised to increase the scope and funding for the project (Public Law 85-162).
October 23, 1957	Site preparation started. An access road was constructed from Highway Route 20.
November 8, 1957	Construction contract for the reactor containment shell awarded.
December 19, 1957	Construction of the containment vessel started.
April 2, 1958	Construction of the Laboratory and Service Building and site development started.
April 14, 1958	Construction of the substation and transmission lines started.
July 1958	Construction of the Power Plant and Reactor Plant started.
April 1959	Construction of Sodium-Boiler Plant and Fuel Cycle Facility started.
June 23, 1959	Argonne National Laboratory moved into the Laboratory and Service Building for first occupancy of the site.
November 10, 1959	Reactor Building rotary bridge crane construction completed.
April 4, 1960	Fabrication of initial enriched fuel loading started.
April 15, 1960	Primary tank installation and installation of the 480 volt and 2,400 volt switchgear completed and energized, establishing the first permanent power.
August 23, 1960	Construction of the main cooling tower completed.
September 1960	Component installation in the Power Plant and Reactor Plant started.
November 1960	Construction of the Power Plant and Reactor Plant completed.

February 1961	*Hazard Summary Report for Dry Critical Experimental Program* (ANL-6299) completed.
April 1961	Fabrication of initial fuel loading completed.
May 12, 1961	Component installation in the Reactor Plant completed.
May 23, 1961	Argonne National Laboratory work preparing for dry critical started in the Reactor Building.
June 14, 1961	Argonne National Laboratory took over component installation in the Sodium-Boiler Plant.
June 19, 1961	Enriched fuel and all equipment needed for approach to dry critical delivered to site.
August 11, 1961	Ten tank cars of sodium arrived at National Reactor Testing Station site.
August 23, 1961	Approval for Dry Critical Program is given by U.S. Atomic Energy Commission.
September 30, 1961	Dry criticality of reactor achieved.
November 3, 1961	Last tank car of sodium arrived at EBR-II site.
December 15, 1961	Component installation in the Power Plant completed. The main turbine operated on 175 pounds steam (plant steam system).
March 7, 1962	The turbine generator synchronized to the National Reactor Testing Station electrical loop for the first time.
April 11, 1962	Elevated temperature test of primary system completed.
June 1962	*Addendum to Hazard Summary Report* (ANL-5719) completed.
July 1962	Filling of primary tank with sodium started.
August 1962	Construction of Fuel Cycle Facility completed.
October 1962	Approach to criticality and wet criticality experiments started.
November 13, 1962	Construction of the Sodium-Boiler Plant completed.
December 26, 1962	All plant construction and component installation completed.
January 29, 1963	First tank car of sodium transferred to secondary sodium storage tank.
February 1963	Component installation in Fuel Cycle Facility completed.
July 1963	“Hot operations” of Fuel Cycle Facility started.
November 11, 1963	Wet criticality of reactor (with sodium) achieved.
December 5, 1963	Wet criticality experiments completed.

April 5, 1964	Combined operation of primary and secondary systems started.
July 16, 1964	Approach to power started.
August 13, 1964	Plant operation at 30 megawatt thermal, turbine generator feeding Idaho National Engineering Laboratory electrical loop at 3 megawatt electrical started.
September 1964	First irradiated fuel processed in Fuel Cycle Facility.
March 27, 1965	Reactor power increased to 45 megawatt thermal.
April 1965	First recycled subassembly installed in reactor.
May 1965	First irradiation experiments installed in the core.
August 26, 1965	Reactor power increased to 50 megawatt thermal.
April 1969	Last recycled fuel subassembly (for a total of 418) installed in reactor.
1969	The mission of EBR-II was reoriented from a demonstration plant to an irradiation facility.
1969	Fuel recycle equipment removed from Fuel Cycle Facility. Facility converted to irradiation examination facility.

APPENDIX B

EVOLUTIONARY PROCESS THAT PRODUCED THE EBR-II CONCEPT

APPENDIX B EVOLUTIONARY PROCESS THAT PRODUCED THE EBR-II CONCEPT

INTRODUCTION

This appendix provides a more detailed description of the evolutionary process through which the EBR-II concept was produced. It is an attempt to respond to the questions: "How did the EBR-II concept evolve?" and "Who invented the EBR-II submerged primary system concept and the fuel cycle?" It is difficult to recreate the processes involved or develop a detailed historical chronology.

This appendix includes historical documents (as attachments) and a description of their significance related to that evolutionary process. By identifying appropriate descriptive material in these documents, and identifying the time at which they apply, the reader should derive an understanding of the evolutionary process and an approximate timeline of the EBR-II concept as it proceeded and matured.

The following documents record activities related to the development of EBR-II and the quest to exploit those unique characteristics. These notes, memoranda, and reports cover the period during which the preliminary concepts evolved into a real facility.

MEETING NOTES: APRIL 26, 1944

The earliest record available consists of considerations of fast reactors for electric power operation in notes from a meeting on April 26, 1944 (ATTACHMENT 1), in which comments by Enrico Fermi and Leo Szilard were noted. These discussions concentrated on the physics of potential power reactor systems, without defining details. The notes list Fermi, Allison, Szilard, Wigner, Weinberg, Seitz, Morrison, Cooper, Vernon, Tolman, Watson, and Ohlinger as attendees. Only the specific remarks of Fermi and Szilard are recorded in the notes (and one aside by Allison).

Fermi apparently opened the discussion with this comment: "It was assumed for today's discussion that the aim of the chain reactor would be the production of power." (Note that "tube alloy" is a code for uranium.) Although the five pages of notes did not include a reactor design description, they did include some interesting observations and predictions, for example:

- "The fundamental aim ... would be to get the maximum possible yield, with full utilization of the metal as a goal... If such a solution is not possible, then the schemes for isotope separation should undoubtedly be investigated further."

- "Another type of pile to consider is one with very little or no moderator (fast chain reacting type).....is simple in principle but, practically, it involves serious problems in recovering the heat."

The notes continue to state that "Mr. Szilard was the second speaker and proposed approaching the problem from a different viewpoint..." (Apparently, Szilard used figures in his presentation, which were not included with the meeting notes.)

- "In Sketch A, the enriched tube alloy (enriched to where the chain reaction will go) and natural tube alloy would be distributed in the form of rods in a cylindrical pile, in which the enriched material would be in the center portion of the rods lying within a circular area in the center of the pile" (emphasis added).

- "The coolant for this type pile would be a bismuth-lead alloy and would flow downward through the pile between the static and rotating rods. The possibility of using liquid sodium in place of bismuth-lead should also be looked into."

The meeting notes contain additional comments that reflect the insight and intuition of the participants in April 1944. They discuss the formation of plutonium-240 in thermal reactors “which we will call super plutonium” (Szilard). He assumes that there is a 50 percent probability that “this new element will be fissionable.”

- “If it is not fissionable, it is assumed there is a 50% chance that it will be formed in negligible quantity in the capture of fast neutrons.”
- Of particular interest were Szilard’s specifications for a reactor. Although he presumed lead-bismuth as the coolant, he noted that sodium has about the same volumetric heat capacity but approximately one-tenth the density. His early vision of a reactor is as follows:
  - “...The enriched core would be about 1/2 to 1 meter in diameter by about the same height.”
  - “...The bismuth-lead alloy would occupy about 1/3 of the enriched core and would pass through the pile at a velocity of about 15 meters per second.”
  - “With 1/2 cm. diameter rods raised to 700°C metal temperature at the center of the central rod and with 150°C temperature increase in the coolant, about 250,000 KW will be removed.”

In addition to these specific visions, these notes include discussion of internal and external (core and blanket) breeding, the contribution of “fast fissions” in uranium-238 to breeding and various techniques for enhancing breeding capability.

The size of Szilard’s reactor is in range, the diameter of the fuel element is comparable to EBR-II, but the coolant flow rate is almost double EBR-II and all of the temperatures are significantly higher. The maximum metal fuel temperature and the high coolant flow-rate would introduce engineering problems, but these physics concepts warrant much appreciation and praise. (These notes were not available to the EBR-II designers.)

The meeting described above took place long before EBR-II, but the notes provide an insight about the early recognition of some of the unique potential characteristics of fast reactors.

Due to the poor quality of the meeting notes dated April 26, 1944, discussing the production of power and breeding from nuclear reactors, only the first page of the original meeting notes is reproduced here but a retyped version of the original notes is also included within this attachment (Attachment 1).

The following key may be useful in understanding the minutes and their intent.

Key		
49 or 9	=	plutonium-239
28 or 8	=	uranium-238
25	=	uranium-235
40-10	=	plutonium-240

MEMORANDUM: L. J. KOCH TO DISTRIBUTION, SEPTEMBER 22, 1952 (ANL-LJK-1)

This memorandum (ATTACHMENT 2) contains a preliminary concept of a full-scale plutonium breeder reactor. The pilot plant would become EBR-II. The plutonium breeder reactor, a "full-size" power station would be 150 to 200 megawatt electric. Of particular significance is the assumption of plate-type fuel elements reflecting the high power density requirement and the implied acceptance of a conventional reactor and piped system concept. However, even at this early date, some special requirements were identified, which would lead to the EBR-II submerged primary system concept, such as rapid refueling, sealing moving parts between a sodium and air environment, and hot leg pumping of sodium.

MEMORANDUM: L. J. KOCH TO DISTRIBUTION, NOVEMBER 10, 1952 (ANL-HE-1529)

This memorandum (ATTACHMENT 3) shows the refined ideas about both the plutonium breeder reactor and the prototype. It established the prototype at 10 percent of the volume and power level of the plutonium breeder reactor resulting in equivalent power density of the two concepts. It also reflects some refinement in details of a plate-type fuel element, but most importantly, reflects the impact of refueling on conceptualization. Attention to this requirement led to consideration of two reactors coupled to one power system. This idea was short-lived, but reflects the challenging requirements of the refueling of high power density reactors. This recognition contributed to the later evolution of the EBR-II submerged primary system concept.

MEMORANDUM: L. J. KOCH TO W. H. ZINN, MARCH 4, 1953

This memorandum (ATTACHMENT 4) defines the performance objectives and requirements related to flexibility and variability of the reactor. It stresses the need for passive shutdown cooling and the need to avoid rapid temperature changes in the inlet coolant and the desirability of providing a large heat capacity in the coolant system (although it is assumed to be best provided in the secondary system).

Perhaps most important is the introduction of a revised development approach. The original plan was to design a plutonium breeder reactor and then scale down the design to provide a basis for the prototype or pilot plant (i.e., EBR-II). It was now proposed to design the prototype as a flexible machine, whose features and experience could be scaled up to produce a plutonium breeder reactor. This changed the emphasis and effort that was then directed to the development of a prototype. The proposal was directed toward a conventional reactor-loop system but with the use of an enlarged reactor vessel to provide some of the flexibility desired.

MEMORANDUM: W. H. ZINN TO DISTRIBUTION, MARCH 9, 1953

This memorandum (ATTACHMENT 5) is a request to organize the development work by Argonne National Laboratory that is related to the plutonium breeder reactor program. It is an early effort to coordinate the various technical development programs that were being pursued in support of fast reactor technology development. Dr. Zinn was beginning to think of this program in terms of a project.

MEMORANDUM: L. J. KOCH TO W. H. ZINN, MAY 28, 1953 (ANL-LJK-4)

This memorandum (ATTACHMENT 6) summarizes all of the work being conducted in the laboratory related to the fast reactor development program. It provides some insight into the breadth of research and development being performed, which might contribute to fast reactor development. Some of this work was being performed as a part of the "basic research" program of the laboratory (i.e., not associated with the reactor development program). Some of the work could provide basic technology which would contribute to the program. Also much of the reactor development program was producing information useful to all types of nuclear reactors including fast reactors. It was important to identify specific information which would be useful to the EBR-II program.

To provide orientation and perspective, the divisions of the laboratory were then coded as follows:

- Reactor Development Divisions
  - RE Reactor Engineering Division
  - CEN Chemical Engineering Division
  - EBR EBR Operations-Idaho
  - RCD Remote Control Division
  - MET Metallurgy Division[a]
- Basic Research Divisions
  - PHY Physics Division
  - CHM Chemistry Division
  - MET Metallurgy Division[a]

MEMORANDUM: L. J. KOCH TO N. HILBERRY, SEPTEMBER 2, 1953 (ANL-LJK-6)

This memorandum (ATTACHMENT 7) identified the technical requirements for the development and construction of an experimental fast reactor power system (a successor to EBR-I). It defined such a reactor, as it was conceived at that time, and described a program to achieve it. Argonne National Laboratory was involved in the development of nuclear power reactors for the generation of electricity and for U.S. Naval propulsion systems. Commercial nuclear powered electricity generation was a high-priority objective of the U.S. Atomic Energy Commission. The laboratory believed that another step in the development progression following EBR-I was necessary before fast reactors could be included as a candidate for this application. Dr. Hilberry requested a description of that next step and the requirements to achieve it. The laboratory played a major role in bringing nuclear power to fruition. The memorandum is self-explanatory and the last major paragraph summarizes the then current thinking about the pilot plant (followed by unrealistic schedule assumptions).

PAPER: *THE ENGINEERING DESIGN OF EBR-II*, AUGUST 1955

This paper was prepared for the first International Conference on the Peaceful Uses of Atomic Energy. This paper is available in the proceedings of the meeting (Barnes et al. 1955).

Excerpts from this paper are included (ATTACHMENT 8) to continue the story about the evolution of the EBR-II concept. They highlight the status of the EBR-II concept as of mid-1955. The most significant aspects of the concept evolution at this time are: (1) the overall plant arrangement; (2) the submerged primary system concept; (3) the use of a direct current electromagnetic pump and homopolor generator for the primary sodium circulation system; (4) the integral process cell; (5) a once through steam generator; and (6) a central blanket. Some specific details were changed significantly, such as the primary sodium flow was provided by one electromagnetic pump for the core and another for the blanket. The reactor cover was raised and rotated to the side to provide access for fuel handling.

A very significant change occurred subsequent to this presentation. As shown, the reactor configuration included tube supports for the upper end of each subassembly, protruding from the reactor cover, which were intended to permit axial expansion of each subassembly but maintain vertical location. This arrangement would provide positioning of each subassembly at the bottom and top. Further evaluation revealed that with the restriction of the radial motion at the top of the subassemblies, inward bowing

a. The Metallurgy Division was conducting basic research programs, as well as reactor development programs.

toward the center of the cove may occur. This bowing would produce a positive component to the power coefficient of the reactor. As a result, the tube supports were removed in the final design.

The changes evolved quite naturally in regard to available technology and/or recognition of better options. As described earlier, the concept of a central blanket was abandoned because of the small size of the EBR-II reactor. The most interesting changes from the concept described in this paper involve the overall plant arrangement. The departure from a single building containing the entire plant to four separate buildings and structures is instructive of an evolutionary process responding to specific requirements.

MEMORANDUM: W. H. ZINN TO A. H. BARNES, JANUARY 4, 1956

This memorandum (**ATTACHMENT 9**) summarizes the requirements established by the laboratory director for proceeding to the next step in the process of achieving an EBR-II facility. It describes the process to be followed to permit the director to proceed in establishing the basis for his decision-making process. A similar memorandum was sent to other division directors.

MEMORANDUM: W. H. ZINN TO L. J. KOCH, JANUARY 9, 1956

This memorandum (**ATTACHMENT 10**) is a charge to the EBR-II Feasibility Evolution Committee from the laboratory director, who also served as chairman of the committee. Similar memoranda were sent to each member of the committee.

This detailed and structured procedure for establishing feasibility of the EBR-II concept is a reflection of the radical approach that this concept represented. Although there was strong support and enthusiasm for the concept and an eagerness to pursue it, there was recognition of the risks involved. The concept represented a radical departure from existing practice.

MEMORANDUM: L. J. KOCH TO W. H. ZINN, FEBRUARY 10, 1956

This memorandum (**ATTACHMENT 11**) describes the results of the EBR-II Feasibility Evaluation by the committee and discusses questions and comments of committee members.

POST FEBRUARY 10, 1956 ACTIVITIES

The EBR-II Feasibility Evaluation accomplished its purpose. It identified the uncertainties and risks associated with pursuing the EBR-II concept and identified the rewards that could be achieved if it was successful. The laboratory elected to pursue this challenge and achieve the potential rewards. This process proceeded in an orderly manner by pursuing the technical development program needed to support the program, and which were verified and clarified during the EBR-II Feasibility Evaluation. The detailed design of the EBR-II concept was pursued in parallel. The H.K. Ferguson Company in Cleveland, Ohio was selected as the architect engineer for EBR-II.

The design features of EBR-II were established to the extent possible. Design features still under consideration were identified, and provisions were made to allow adjustments and changes in the design to reflect relevant technological information as it became available. This strategy impacted the construction schedule and complicated the construction effort that were being performed under fixed price contracts.

Except for those special considerations, the design and construction program proceeded in the traditional manner of a complex technical project. The involvement of an architect engineer in the project stimulated concentration on the design of a real plant. The cost of this real plant was estimated and a realistic cost was incorporated into the laboratory's request to the U.S. Atomic Energy Commission for authorization of the EBR-II project. The project was authorized by Congress in August 1957 (but requiring funds appropriation) and formally authorized by the U.S. Atomic Energy Commission (including funds) on March 3, 1958 (**ATTACHMENT 12**).

ATTACHMENT 1

MEETING NOTES: APRIL 26, 1944

REPRODUCTION OF THE FIRST PAGE OF ORIGINAL NOTES

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[redacted]TES ON MEETING OF APRIL 26, 1944

9:00 - 10:30 ACK-209 N-1729w

Pres[redacted]t: Fermi, Allison, Szilard, Wigner, Weinberg, Seitz, Morrison, Cooper, Vernon, Tolman, Watson, Ohlinger

The first speaker in today's meeting was Mr. Fermi. His remarks follow.

It was assumed for today's discussion that the aim of the chain reaction would be the production of power.

The first type of pile assumed for this purpose was a permanent large pile of about the Hanford size (but not the Hanford type necessarily) for production of energy in the neighborhood of 10^6 kilowatts. The arrangement suggested was one in which one large mother plant would produce 49 for consumption in a series of smaller plants. In the mother plant, the energy produced would be used to reduce the cost of the 49 produced. (Mr. Fermi mentioned that he viewed the use of this power for the heating of cities with sympathy). There may be non-technical objections to this arrangement, for example, the shipment of 49 to the smaller consuming plants offers the serious hazard of its falling into the wrong hands, but these were to be omitted from this discussion.

The fundamental aim in the mother plant would be to get the maximum possible yield, with full utilization of the metal as the goal. If a solution to such a proposal can be found, then the schemes for isotope separation are not of great interest. If such a solution is not possible, then the schemes for isotope separation should undoubtedly be investigated further.

In the following discussion of full metal utilization, the isotopes 28 and 49 will be referred to as 8 and 9, respectively. In the reaction cycle suppose that one fission of 9 and ψ fissions of 8 take place in a single cycle or generation. Then the production of neutrons will be $\nu_9 + \psi\nu_8$. Some neutrons are lost in the moderator, coolant, etc. Let L = the number lost and α = the number used in producing 40-10. Then the excess of neutrons available for absorption by 8 to produce 9 will be

$$(1 - L)(\nu_9 + \psi\nu_8)$$

RES[redacted]ED DATA

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and the production of 9 per cycle will be

$$(1 - L)(\nu_9 + \psi\nu_8) - 1 - \alpha - \psi .$$

The term $1 + \alpha$ represents the destruction of 9. Therefore, the ratio of production of 9 to its destruction, which we will call γ, will be

$$\gamma = \frac{P}{1+\alpha} = (1 - L)\left(\frac{\nu_9}{1+\alpha} + \psi\frac{\nu_8}{1+\alpha}\right) - 1 - \frac{\psi}{1+\alpha}$$

[redacted]

RETYPED VERSION OF THE ORIGINAL NOTES

NOTES OF THE MEETING ON APRIL 26, 1944

Present: Fermi, Allison, Szilard, Wigner, Wienberg, Seitz, Morrison, Cooper, Vernon, Tolman, Watson, Ohlinger

The first speaker in today's meeting was Mr. Fermi. His remarks follow.

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$$(1-L)(\nu_9+\Psi\nu_8)$$

and the production of 9 per cycle will be

$$(1-L)(\nu_9+\Psi\nu_8)-1-\alpha-\Psi$$

The term 1 + α represents the destruction of 9. Therefore, the ratio of production of 9 to its destruction, which we will call γ, will be

$$\gamma = \frac{P}{1+\alpha} = (1-L)\left(\frac{\nu_9}{1+\alpha}+\Psi\frac{\nu_8}{1+\alpha}\right)-1-\frac{\Psi}{1+\alpha}$$

To utilize all the metal, γ obviously must be greater than 1. If γ is only very little greater than 1, the chain-reaction would keep going with maximum economy of fissionable materials and would continue to go on until all the metal were used, but the value of such a pile would not be great and it would only be good for, say, hardening materials (the Wigner effect) or possibly (through less desirable) heating cities. The effective ν_9 is around 2.1 to 2.2.

Assume first a Hanford type pile with an equivalent amount of 49 substituted for the 25, i.e., in the early stages, 25 would be burned to produce 49 which would gradually improve its condition. The earlier estimate of 1.9 for the ratio of the fission cross section of 49 to that of 25 has been more recently estimated by Y as 1.4. The ratio of absorption cross section for 49 to that of 25 is around 1.5. With these conditions, ν_9 is about 10% higher than it was previously thought to be. (The actual values of ν and ν effective are not really known so the discussion can only show ranges.) The situation then in a pile of Hanford design and lattice would be for a ν effective (which will be referred to hereafter as μ) of 2 to 2.2, γ will be from 0.8 to 0.98. In the latter case, the pile is close to a balanced situation but not quite there. To adjust such a pile without drastic changes of design, large diameter slugs or more metal could be used to improve the thermal utilization and increase Ψ. However, over-sized lumps increase the difficulty of cooling since the annular type cooling is badly limited in power production by the metal temperature.

The second type pile considered for the production of power was the P-9 moderated pile. For a μ of 2 to 2.2, γ would be 0.93 to 1.13. These values do not necessarily represent the optimum but are merely indicative of what can be done with P-9 piles and one with such a γ of 10 to 15% may or may not be an operable plant. The practical difference between continuous and discontinuous P-9 plants is not large in this respect since the loss by absorption for the coolant and its tubes practically compensates for the less efficient reproduction in slurry piles. One might hope to improve the situation by capturing the escaping neutrons in a reflector but the absorption in the pile container is an important problem.

Another type of pile to consider is one with very little or no moderator (fast chain reacting type). From the nuclear plant of view this is very desirable and is simple in principle but, practically, it involves serious problems in removing the heat. Ignoring the cooling, and considering only the nuclear plant point of view, this type pile may be of either one of two forms:

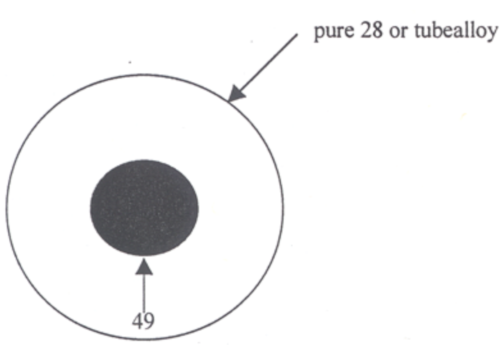

Fig. A

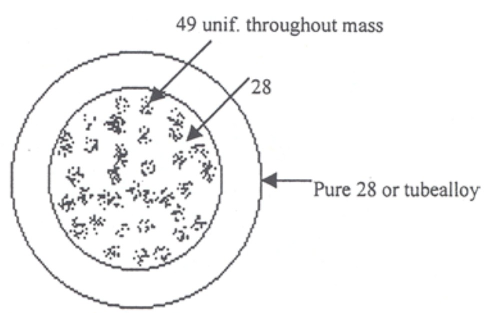

Fig. B

In Fig. A, a small spherical core of 49 say, 10 cm in diameter, would be surrounded by a sphere of 25 or normal tubealloy 40 to 60 cm in diameter. This arrangement is good from a γ standpoint and one might expect a γ of 1.3 to 1.4, because L can be made small since the fast neutrons from the 49 get into the 28 readily. (Mr. Allison pointed out that if 25 is not considered for the surrounding here, thorium might be used). The pile shown in Fig. A only requires a few kilograms of 49. To utilize more 49 it would be possible to construct units like A with multiple 49 cores spherical or cylindrical in shape.

Fig. B represents a homogeneous sphere of 28 with 49 uniformly distributed throughout the mass, the whole surrounded by a reflector of pure 28 to catch the leakage neutrons. In this arrangement about 70% of the neutrons get into 28 immediately to produce fast fission. Assuming a mixture of 49 and 28 in which X represents the percentage of 49, critical conditions

(i.e., where the chain reaction continues if the pile is of infinite size) would be reached with about 5% of 49 in the mixture (X=0.049). For values of μ of 2 to 2.2, γ would be 1.37 to 1.57. As the pile size is decreased, the following results would be obtained. They are calculated without reflector.

<u>Table I</u>

Critical Radius of Sphere	X (fractionof 49)	γ $\mu = 2$	γ $\mu = 2.2$
100 cm	0.056	1.23	1.43
70 cm	0.060	1.10	1.30
50 cm	0.067	0.98	1.18

Adding a reflector would decrease the critical radius of the active sphere by about 10 cm and improve very considerably the value of γ since the reflector would utilize the neutrons escaping from the active core. Taking the core to the 70 cm sphere above, this represents about 1 ½ m^3 or say 30 tons of the mixture. Therefore, 6% or about 2 tons of 49 would be required to keep this machine running. Thus a plant of this type requires a large quantity of 49 for operation although this is not sufficient reason for discarding this type of pile as a possibility.

<u>The serious</u> objection to <u>these</u> fast <u>chain</u> piles <u>is</u> the <u>removal</u> of the <u>heat</u>. Since practically <u>all</u> the heat is produced in the 49 (about 70 to 80%), piles like those in Fig. A are harder to cool since it is mainly <u>the tiny</u> core which <u>must be cooled</u> while in Fig. B the whole mass is to be cooled.

As another possibility, a compromise enriched pile might be designed which would have enough moderator to reduce the percentage of enrichment required to keep the chain reaction going. But not as large an amount would be required for the conventional optimum conditions.

Mr. Fermi suggested that at a later meeting he would consider question of how to use the 49.

Mr. Szilard was the second speaker and proposed approaching the problem from a different viewpoint, — that of assuming more optimistic values of the constants so as to indicate other potentialities. He pointed out that the fast reaction is preferable to the slow chain reaction for producing 49 from tubealloy and that this is probably more true if we assume more pessimistic values for ν or μ. Before discussing these values of the constants, sketches of a possible design were distributed and described briefly. These sketches are attached hereto.

The sketches show two different arrangements. In sketch A, the enriched tubealloy (enriched to where the chain reaction will go) and natural tubealloy would be distributed in the form of rods in a cylindrical pile, in which the enriched material would be in the center portion of the rods lying within a circular area in the center of the pile. Part of the rods, located within three circular areas around the center (as indicated in Fig. 1) would be arranged so the cylindrical bundles could each be rotated about its axis. In each of the rotating bundles, part of the rods would be natural tubealloy and the balance of natural tubealloy with the center section enriched.

In the beginning, the enriched material in the three bundles would all face the center of the pile and lie within a cylinder whose axis would coincide with the axis of the pile and whose cylindrical surface would pass through the three axes of the revolving bundles. By means of this arrangement, as the multiplication factor increased with the continued operation of the pile, the enriched material could be rotated away from the center of the pile and the tubealloy brought towards the center where it in turn would be enriched. In the center of the pile would be a single tube for introducing mercury, liquid bismuth, or some other absorbing or slowing material for controlling the pile. The coolant for this type pile would be a bismuth-lead alloy and would flow downward through the pile between the static and rotating rods. The <u>possibility of using liquid sodium in place of bismuth-lead should also be looked into.</u> The volumetric heat capacity of the liquid sodium is about the same as that of the bismuth-lead alloy but its density would be 10 times less, so that the pressure drop would be about 1/10 that for the bismuth-lead alloy or the velocity about 3 times larger for equal pressure drop. In the scheme just described, the following approximate conditions would obtain: (1) the bismuth-lead alloy would occupy about 1/3 of the enriched core and would pass through the pile at a velocity of about 15 meters per sec; (2) with 1/2 cm diameter rods raised to 700^{o}C metal temperature at the center of the central rod and with 150^{o}C temperature increase in the coolant, about 250,000 kw will be removed. The pumping power for the coolant will consume about 5% of the power produced.

In the alternate scheme B, control of the pile would be obtained by means of a nest of tubes for the mercury or other controlling medium arranged as in Figs. 3A and 3B and 4A and 4B. The metal rods would all be stationary and vertical (nos. 12, 13, and 14 in Fig. 3A) and would be about ½ to 1 cm in diameter by about 2 meters long.

In both designs, the enriched core would be about 1/2 to 1 meter in diameter by about the same height. The balance of the material around the core would be ordinary tubealloy of the same rod size. The total diameter and the height of the pile would be about 2 meters.

The objective of such a pile must be to produce as much extra 49 as invested. It is assumed that the production will be double the original investment. For every atom of 49 disintegrated, two atoms of 49 would be produced. Part of these will be produced in the enriched core and part in the surrounding natural tubealloy. Some of the production in the core will tend to leak out into the natural tubealloy and this leakage must be kept within certain limits. Then k will increase over a period of time. As the chain reaction goes on, the multiplication factor k will then increase so that the controls must provide for this as well as the normal operating control of the pile.

In the slow chain reaction, 49 captures neutrons in radiative not fission capture to produce a new element which we will call super plutonium or 40-10. It is assumed there is a 50% chance that this new element will be fissionable. If it is not fissionable, it is assumed there is 50% chance that it will be formed only in negligible quantity in the capture of fast neutrons. Thus, there is a 75% chance in a fast chain reaction that we may use ν and not μ in getting the production balance (μ = 2.2 neutrons per neutron absorbed, ν_{25} = 2.2 x 1.175 = 2.6 neutrons produced per neutron absorbed). As the energy of the neutrons increases from thermal to fission energies, it is assumed there is no decrease in ν. The main argument in favor of the fast chain reaction is that if a fission neutron is released in tubealloy, it causes fission in the 28 to produce 1.2 neutrons (fast effect). If all the neutrons are captured, the overall balance would be that for every atom of 49 destroyed, two atoms of 49 would be produced. One goes back into the chain reaction, the other replaces the 49 destroyed, providing a net gain in 49.

In experiments in which Ra - Be (?) neutron source was surrounded by 28, measurements indicated a 5.3% increase in the number of neutrons and that 63% of the neutrons remained above the fission threshold. This means that the increase in the number of neutrons for an infinite sphere would be

$$\frac{5.3}{1 - 0.63} \text{ or } 19\ 1/2\%.$$

If the fission cross section is taken at 0.35 and the inelastic cross section at 2.7 for a ν_{28} of 2.2 to 2.6 ε will vary from 1.18 to 1.245.

Referring to the value above of ν_{25} of 2.6, if we were to use the more optimistic results reported by Y (that ν_{49} is 20% larger than ν_{25}) then ν_{49} equals 3.1 neutrons produced per neutron absorbed. If we are less optimistic and assume ν_{49} effective = 2.5 but use the 19 1/2% increase indicated by the experiment mentioned above, we have three neutrons produced in a mixture of 28 and 49 for one atom of 49 destroyed.

It has been suggested that one of the subjects for one of the meetings soon to be held would be a review of the availability of the metal producing ores and other sources of tubealloy.

This is to be given by Mr. P. Morrison.

ATTACHMENT 2

MEMORANDUM: L. J. KOCH TO DISTRIBUTION, SEPTEMBER 22, 1952 (ANL-LJK-1)

9/22/52

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~~Reactor Engineering Division~~

102

To: Distribution

From: L. J. Koch

Reactor Engineering Division

9-22-52

Subject: PBR Preliminary Specifications and Considerations

The design of a Pilot Plant for a Plutonium Breeder Reactor will require a prior preliminary study of the full size machine, to establish the feasibility and general configuration of this reactor. The Pilot Plant can then be designed to most effectively contribute to the design, construction, and operation of the Reactor. A very preliminary consideration of a Plutonium Breeder Reactor is presented here in the attempt to establish a basis for study and evaluation, which will define some of the basic problems and indicate the effects of some of the many variables on the overall characteristics of the Reactor.

It is expected that the PBR will be liquid metal cooled fast reactor of approximately 500,000 KW power rating, and be a net producer of electrical power. It will be compact in design and must meet production type requirements of reliability, ease and speed of reloading, and safety. Experimental facilities and flexibility of operation will not be primary considerations.

The coolant will be sodium or sodium-potassium alloy. The core will be small with high heat transfer type fuel elements; probably of plate or pin type. A preliminary specification is attached which is based on Na as the primary coolant, and plate type fuel elements. These specifications should be considered as only approximate and primarily intended to indicate the relationship of the variables and to indicate the type of operating characteristics to be expected.

Core and Fuel Element:

Sketches of a tentative core configuration and fuel element design are attached. Although these elements do not conform exactly with the assumptions used in the specifications, they do indicate the feasibility of these assumptions at this early date.

Because of the compact design of the core, (the fuel elements will essentially be in contact with each other) it may be difficult or impossible to provide an upper grid structure for the support of the fuel elements. It would be advisable to consider the feasibility of a design in which the fuel elements are self supporting at the upper end. An alternate approach, of course, would be a removable grid structure. The former approach appears preferable because the fuel elements must be supported during partial reloading, and it would be desirable to have them self supporting. Ease and speed of

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reloading are primary considerations, and continuous reloading should be considered, but it appears that this will be much more difficult to achieve, and probably will not be an operational requirement.

The core material will probably be a plutonium alloy (if available) with the diluent to be normal uranium if an internal breeder is desired, or a stabilizing element if internal breeding is not desired. The composition of this alloy will be determined by the nuclear characteristics desired and the considerations of breeding gain, critical size, burn-up, etc. A tentative maximum temperature of 600°C has been established for the core material. After the desired alloy has been determined, an extensive investigation must be made to evaluate this limitation and determine the feasibility of increasing this temperature. The clad and structural materials have not been established, and all possibilities should be investigated for there is considerable flexibility of materials in this type reactor. Basically the three requirements are: sound bonding to the core material, stability in sodium at high temperature, and nuclear stability. The cost of the fuel element will be a major consideration, for the economics of the fuel element fabrication are an important factor in the economic analysis of the breeding cycle.

Breeding Blanket:

The general configuration of the breeding blanket has not been defined as yet. The desired nuclear characteristics must first be determined, and then these features incorporated with regard to unloading, cooling, and control, etc. Such things as composition, density, moderation (if any), heat generation, and length of cycle in the blanket should be investigated as soon as possible.

Control:

It appears that reactor control may be achieved by movement of: core material, reflector, blanket, or a combination of these. The feasibility of each method should be considered and will require close cooperation between the physics and design groups. Large amounts of excess reactivity are not required, because burn-up and temperature effects are the only major considerations. The control should be achieved with a minimum perturbation of the neutron flux distribution because of the desire for maximum nuclear effeciency.

Coolant and Flow System:

It is anticipated that the primary coolant system will contain sodium, that the heat will be transferred to a secondary system containing NaK and then utilized in a boiler to produce steam for a rather conventional power plant. (It should be noted, however, that a secondary system is not essential to the system) Primary emphasis will be on the primary coolant circuit as it involves the reactor and its components. A single pass system flowing through the reactor from bottom to top is being considered. This does not exclude other arrangements, for this is one of the most critical considerations in the overall design. Because of mechanical considerations

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it could be desirable to have a free gas space at the top of the reactor for unloading and control functions. Mechanisms may be desired within the reactor vessel, or motion through the reactor lid requiring adequate seals may be required. Obviously sealing against the blanketing gas is far simpler than sealing to the atmosphere against sodium. Maintaining a low head or low level of coolant in the tank will require a low pressure drop in the outlet circuit from the reactor. This may best be achieved by placing the heat exchanger after the pump in the primary circuit. This presents the undesirable condition of pumping the hot fluid, and imposing additional temperature requirements on the pump. A decision of this type can be made only after sufficient information is available to weigh the advantages and disadvantages as a whole, but this factor should be considered as a possible pump requirement.

The attached preliminary specifications indicate the general design characteristics of the reactor, and should serve as a starting point in evolving a feasible reactor design. In no way should these figures be considered as fixed or frozen, nor implied as specific design objectives. The assumptions of core composition are probably not too unrealistic but will undoubtedly change. The coolant temperature rise might quite well be unrealistic with regard to maximum core metal temperature and stresses. These preliminary specifications are intended to present a general picture of the problem, and to permit the various groups to evaluate and augment them as they apply to their special problems and considerations. As this information is evolved and more of the basic parameters are established, a more definite specification and program may be prepared.

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1.

PBR Preliminary Specifications and Considerations
(Core and Primary Coolant)

Core:

Size:

36" dia. x 36" long.
Core Area = 7.0686 sq. ft.
Core Volume = 21.2 cu. ft.
Composition: (by volume)
20% Active Mat'l. (Pu diluent) — 30%
20% Structural Mat'l. (Fe) — 15%
60% Coolant (Na) — 55%

Power:

500,000 KW (core power)
.20 x 21.2 cu. ft. = 4.24 cu. ft. core material. 6.36 ft^3
$\frac{500{,}000}{4.24}$ = 117,900 KW per cu. ft. of core mat'l. 78,500
$\frac{117{,}900}{523}$ = 225 KW per Kg of core mat'l. 150 ~~(specific power)~~
21.2 x 28.32 = 600~~0~~ liters.
$\frac{500{,}000}{600\cancel{0}}$ = 83~~0~~KW per liter (power density)

185
62.4
11544
2.2 = 523 Kg/ft^3

Fuel Element: (Plate Type)

.040" thick core mat'l. — .061"
.010" thick clad (each side) .007
.060" total plate thickness. .075
.080" coolant channel. .097

Heat Flow:

$\frac{4.24 \times 12}{.040}$ = 1272 sq. ft. of core mat'l. $\frac{6.36 \times 12}{.061}$ = 1250 sq ft.
1272 x 2 = 2544 sq. ft. effective heat transfer area. = 2500 ft^2
$\frac{500{,}000 \times 56.89 \times 60}{2544}$ = $.67 \times 10^6$ BTU/sq.ft.hr Avg. Heat Flux. $.68 \times 10^6$

Coolant: Flow and temperature rise:
(assume 50% of area for effective flow)
.50 x 7.0686 = 3.534 sq. ft. flow area
25 fps flow velocity.
3.534 x 25 = 88.35 cu.ft./sec. flow rate 39,657 gpm
@ 850°F - sodium
C = ~ .304 BTU/lb.°F
ρ = 52.5 lbs./ft.^{3}

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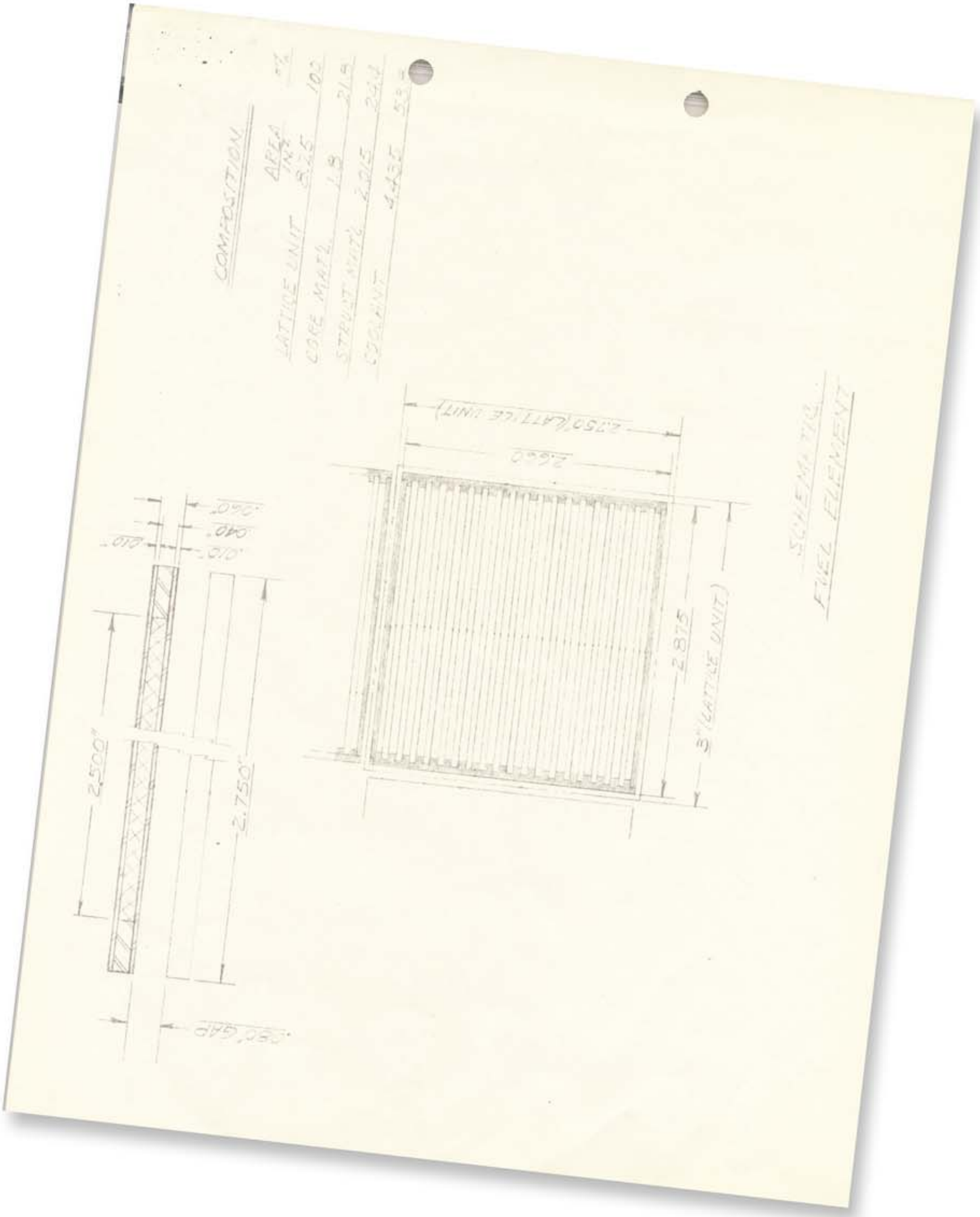
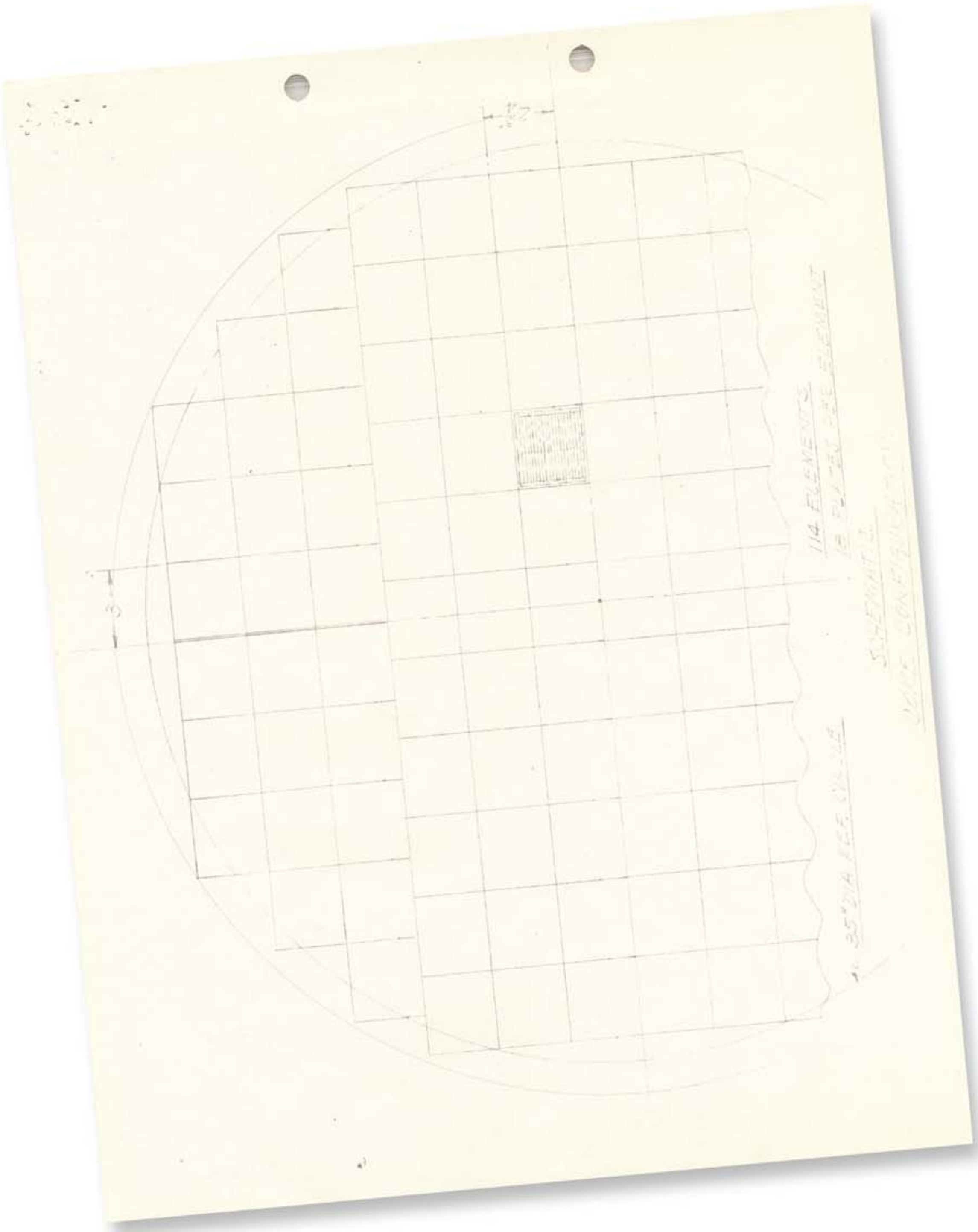
88.35 x 52.5 = 4638 lbs./sec.
4638 x .304 = 1410 BTU/sec.° F
500,000 x $\frac{56.89}{60}$ = 474,000 BTU/sec. total heat generation ~~(avg.)~~

$\frac{474,000}{1410}$ = 336° F temperature rise. (avg.) 9.33°F per inch of length



COMPOSITION
AREA IN²
%
LATTICE UNIT 8.25 100
CORE MAT'L 1.8 21.8
STRUCT MAT'L 2.015 24.4
COOLANT 4.435 53.8
2.750 (LATTICE UNIT)
2.660
SCHEMATIC FUEL ELEMENT
2.875
3" (LATTICE UNIT)
.060"
.040"
.010
.010"
2.500"
2.750"
.080" GAP


SCHEMATIC
CORE CROSS SECTION
COOLANT LEVEL
SUPPORT ZONE


114 ELEMENTS
35" DIA EFF. CIRCLE
3"

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MEMORANDUM: L. J. KOCH TO DISTRIBUTION, NOVEMBER 10, 1952 (ANL-HE-1529)

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102

November 10, 1952

To: Distribution

Department: Reactor Engineering

From: L. J. Koch

Subject: PBR Preliminary Specifications and Considerations

The present tentative plans for a Power Breeder Reactor include a conceptual design study to establish the feasibility and general configuration of this reactor, followed by a more detailed study of a scaled down pilot plant version of this reactor together with an analysis of the feasibility of first constructing such a pilot plant. This memorandum will serve to summarize the thinking and discussions concerning this subject, and to establish a basis for the program of study and evaluation required. It is necessary also to define as early as possible some of the basic problems and considerations affecting the characteristics of these reactors.

Basic Design Considerations:

To provide a basis for the evaluation of these reactors, some preliminary specifications have been established which will serve to correlate the efforts of the various groups concerned with these reactors. Appended to this memorandum is a summary of these preliminary specifications for a Power Breeder Reactor, and a Pilot Plant, together with the comparable specifications for EBR where applicable.

The PBR has been assumed to be a 500,000 KW core power fast reactor, sodium cooled, with a core volume of 600 liters. The Pilot Plant may be scaled down in many ways, but will inevitably result in the compromise of some basic characteristics of the Pilot Plant as compared to the PBR. A Pilot Plant is considered here which is rated at 10% of the power rating, and 10% of the volume of the PBR. The power density of the two machines is therefore the same, and results in the scaling up of these reactors from the EBR by factors of 10 in volume; namely, 6, 60, and 600 liters respectively. The power density of EBR, as well as many other basic characteristics, is not comparable to those for the proposed machines.

Core and Fuel Element.

A core composition for the PBR has been assumed, based on very preliminary nuclear considerations of the reactor. These present assumptions provide that the dilution may be varied to adjust the core

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composition as the calculations are further refined. This approach permits establishing tentative physical dimensions for the fuel elements in accordance with the assumed core composition and thereby determine the thermal characteristics of the reactor. A fuel plate type element of rather conventional design has been assumed for this purpose, and the dimensions of such an element are given in the preliminary specifications. Fuel element considerations are not limited to plates; pins are also being considered, and other configurations will also be reviewed if applicable.

Because of the necessity for compact design of the core, and the desire for maximum utilization of this space, the feasibility of a reactor core, without the conventional grid structure is being investigated. It is desired to accomplish this, and yet have the individual fuel elements removable. This approach will result in essentially a completely uniform distribution of the core constitutients throughout the core.

Core unloading poses a somewhat difficult problem to analize at this early date, because these requirements are so closely tied to the overall philosophy of operation of the PBR, both as a producer of fissionable material and a net producer of power. Speedy unloading can be achieved, but obviously the time allotted will have a direct bearing on the cost of the equipment and process. Power demand considerations may require that the plant may be shut down for only a few hours if at all. Unloading and reloading in such a short time will impose the problems of shut down cooling during removal, the handling of the sodium on the elements which would be very radioactive as well as chemically active, in addition to the conventional problems associated with remote handling of very active materials. In anticipation of the rather formidable problems posed by this phase of the design, several approaches are being considered. One approach involves the internal transfer and storage of the entire core assembly until such time that the individual fuel elements may be conveniently removed. A modification of this idea utilizes the reactor as the storage facility, in which case two reactors are required. The latter would provide an arrangement which would potentially always be available to satisfy the power demand, but would obviously require judicious utilization of the auxiliary reactor equipment to minimize duplication and to minimize the overall cost of the plant. Obviously the general philosophy of unloading must be thoroughly investigated to achieve a realistic approach that also satisfies the overall reactor operational requirements. The Pilot Plant, which is not a production machine, does not pose these problems, but should include a similar method of unloading to gain experience which will contribute to the operation of the PBR.

It is necessary in this early stage to consider both plutonium and uranium-235 as fuel. Present information indicates that the plutonium-28 breeding cycle is more efficient than the 25-28 conversion cycle, but economic considerations as well as availability and demand for materials could alter this picture. To provide a base point for the study of the thermal characteristics of the reactor, the transition temperature of the core material was established as a maximum. Preliminary investigation of

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the cycle, indicates that a reasonable steam cycle may be achieved in an all uranium core, but the lower transition temperatures of the plutonium-uranium alloys presents a severe handicap in achieving a desirable overall plant cycle. It is apparent that the assumed limitation of operating below the alloy transition temperature imposes severe limitations on a system which is otherwise capable of higher temperature and thus more efficient operation. The importance of this consideration indicates the need for considerably more information regarding this phenomena of phase change, its effects on reactor design and operation, and methods of circumventing this limitation if, in actuality, it exists.

In the first appraisal of a PBR, the core and the fuel element appeared to present one of the most difficult problems to be resolved. Further analysis and investigation has verified this opinion, and indicates the need for considerable effort before an acceptable core structure will be evolved.

Breeding Blanket:

It has been assumed that the PBR will be an internal breeder in that some of the breeding will be achieved in the core, and the remainder of the conversion will be accomplished in a fertile blanket surrounding the core. The general configuration of the blanket has not been defined as yet, except for a few conceptional designs which may be of assistance in preliminary analysis of the reactor system. The desired nuclear characteristics must be determined, and then these features incorporated with regard to size, unloading, cooling, etc. The physical design of the blanket will be directly affected by such considerations as composition, density, moderation (if any), heat generation, and length of operation cycle.

Control:

It appears that reactor control may be achieved by movement of: core material, reflector, blanket, or a combination of these. The advantages of the fast reactor in that it is relatively insensitive to fission product poisoning, becomes somewhat of a disadvantage in consideration of reactor control, for there are no really effective absorbers. It appears that control must be achieved by movement of fissionable or fertile materials, or by a system which affects neutron distribution such as reflector control. The desire for maximum nuclear efficiency precludes the use of controlled leakage from the reactor as an efficient reactor operating control, but this method may prove attractive as a safety provision for reactor shut down. The reactor control system is of course closely allied with the reactor configuration, and must evolve with it.

Heat Removal System:

It has been assumed that the reactor coolant will be sodium, for its heat transfer properties are sufficiently attractive to offset the basic disadvantage as compared to NaK of higher melting point (97.8C). It is anticipated that the heat will be removed through a secondary

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system containing NaK and then utilized in a boiler to produce steam for a rather conventional power plant. It should be noted that while a secondary system is not essential to the system, the use of such a system, which is not radioactive, and which is chemically compatable with sodium, appears very desirable. The use of a secondary system of NaK also provides a very reliable method of putting heat <u>into</u> the primary system through the heat exchanger to maintain the sodium above the freezing temperature. This is an important consideration for the entire primary sodium system must be maintained above the freezing temperature, and the secondary system provides one such reliable heat source. There are also some obvious advantages in avoiding the proximity of radioactive sodium and water in a boiler, which is accomplished by the use of a non-radioactive secondary system. The boiler construction cost and maintenance costs should also be appreciably lower. These questions can only be resolved after further analysis of the entire plant, and after establishing a desired philosophy of construction and operation.

<u>The PBR vs. The Pilot Plant:</u>

The scale down of the PBR on the basis of equal power density results in a Pilot Plant which duplicates the thermal operating characteristics of the large machine. The average heat flux, and average linear temperature rise of the coolant through the reactor core are comparable, but the specific power is lower. Maintaining the same power density and core composition (except enrichment) will reduce the internal breeding (if any) and reduce the burn up as compared to the PBR. If the linear temperature rise of the coolant through the core is maintained the same, then the proportional pumping capacity of the Pilot Plant is greater, with a lower system temperature differential. This need not be a primary consideration, however, because experience in pumping large quantities of coolant is needed, and plant efficiency is not a primary requirement of the Pilot Plant. If the PBR utilizes four circuits in the primary system of approximately 10,000 gpm each, one such system would be applicable to the Pilot Plant, with the resultant advantage of testing prototype components in the system. Development of pumps, valves, heat exchangers, instruments, etc. would be applicable to both machines. That the Pilot Plant utilizes larger equipment than might be necessary if it were not associated with the PBR is probably not a vital consideration.

The average neutron flux in the PBR has been roughly estimated to be between 5×10^{15} and 10^{16} depending on critical size, dilution, fuel etc. With the same power density, the flux in the Pilot Plant will be in a comparable range, being dependant upon the same variables, which will not be identical in the two reactors. Obviously these reactors will be very valuable irradiation facilities, and emphasis should be placed on making the Pilot Plant such a facility. The PBR being a production installation will probably be limited in this respect.

Since operating temperature limitations may not be completely resolved, except in an operating reactor, the Pilot Plant should be designed to permit operation through a wide temperature range, below, through, and above the transition range. This will undoubtedly complicate

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the heat removal system, and impose some limitations on the steam cycle, but again efficiency in the Pilot Plant is not of primary importance.

It would be very desirable to have sufficient flexibility in the Pilot Plant to permit changing the core configuration, type of fuel element, or constituents of the core material. If a gridless core structure is developed, it does not appear to be too unrealistic to think of the entire core structure being removable and replaced by a different configuration. The Pilot Plant must be considered not only as a prototype for the PBR, but also the facility to advance the art of this type reactor.

It is expected that the numbers and relationships indicated here will change as the analysis of these reactors continues. It is planned to issue future memoranda to revise and review the considerations as additional information evolves.

L. J. Koch

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6.

Preliminary Specifications for Reference Design ~~and Considerations.~~
PBR and ~~Pilot Plant~~ Prototype, (with EBR comparison as applicable)

	PBR	~~Pilot~~ Prototype	EBR
Core Dimensions			
Size - in.	36 dia x 36 lg	16.7 dia x 16.7 lg	7.5 Hex x 7.5 lg
Area - Sq. ft.	7.07	1.52	.34
Volume - cu. ft.	21.2	2.12	.21
Volume - liters	600	60.0	6.0
Core Composition: (by Volume)			
Core Mat'l. (meat) - %	30%	30%	48%
Structural Mat'l.-%	15%	15%	14.8%
Coolant - %	55%	55%	37.2%
Core Mat'l - cu.ft.	6.36	.636	.101
Core Mat'l - Kg.	3,326 500 Kg 25' 300 Kg Pu	332.6	52.8
Core Power:			
Total - KW	500,000	50,000	1,000
KW per Kg of Core Mat'l.	150	150	18.9
Power Density - KW per liter	830	830	167
Fuel Element: (Plate Type):			(Rod Type)
Core Mat'l. - in.	.061 thick	.061 thick	.364 dia.
Clad thickness-in.	.007	.007	.022
Total Element Dimensions.	.075 thick	.075 thick	.448 dia.
Coolant channel-in.	.097	.097	---
Cooling Surface - Sq. ft.	2500	250	13
Avg. Heat Flux. BTU/sq. ft. hr.	.68 x 10^6	.68 x 10^6	.21 x 10^6
Heat Removal: (50% effective flow area assumed)			
Coolant	Na	Na	NaK
Flow Area - sq.ft.	3.534	.76	.10
Flow Velocity -fps	25	25	6.7
Flow Rate - cu. ft./sec.	88.35	19	.67
Flow Rate - lbs./sec.	4638	998	33.4
Flow Rate - gpm.	39,657	8,528	300
Heat Generation BTU/hr.	1700 x 10^6	170 x 10^6	3.4 x 10^6
Avg. temp. rise °F	336	156	137
Avg. temp. rise °F per inch	9.33	9.36	18.3

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ATTACHMENT 4

MEMORANDUM: L. J. KOCH TO W. H. ZINN, MARCH 4, 1953

ARGONNE NATIONAL LABORATORY

March 4, 1953

To: W. H. Zinn — Laboratory Director

From: L. J. Koch — Reactor Engineering

Subject: PBR - Prototype

Introduction

The need for a Prototype in the PBR program has been pretty well established, and the general requirements for such a plant have been somewhat defined. It now appears necessary to establish more definite specifications for this experimental plant.

The Pilot Plant has been defined as a scaled-down version of the PBR, probably on the basis of equal power density, but with much greater operational flexibility. It has been generally agreed that a feasible approach would be to prepare the conceptual designs of the PBR, and establish the basic specifications for the PBR, and then use this information as the basis for the design of the Prototype. It now appears that the feasibility study of the Prototype should be made, before little more than a superficial analysis can be made of the PBR.

This memorandum has been prepared for the purpose of establishing preliminary requirements for the Prototype, and to suggest a design philosophy.

Reactor

It has been assumed that the preliminary specifications and considerations listed in ANL-HF-1529 provide a suitable basis for evaluating the Prototype. Using this as a starting point it should be possible to develop the specifications for a unit which will meet these requirements, and also achieve the desired flexibility. At this time it is more logical to define the problems, or objectives, than it would be to establish the solutions. The solutions implied here are intended only to more clearly describe the problems, and possibly to indicate an avenue of approach to the solution.

An idealized reactor configuration for the purpose of flexible operation and evaluation, as described herein, would probably permit an infinite variety of core and blanket configurations, and permit infinite variation in reactor operating conditions. Obviously we must settle for less, and defining how much less represents the problem at hand. Our objective must be to achieve the maximum flexibility possible, consistent with good judgment and economic limitations.

The reactor proper, overly simplified, can probably be considered to be a large shielded vessel with provisions for flowing coolant through it. The degree of flexibility to be achieved in loading the reactor will be dependent upon the degree of flexibility which can be built into the loading and unloading process, and the internal construction of the reactor. It is proposed here that

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the reactor vessel be a relatively large tank, possibly 8 or 9 feet in diameter, with a top shielding plug providing access to any part of the vessel. Although this is a considerably larger vessel than would be required for an experimental plant as presently conceived, it would provide available space outside the blanket for instrumentation, and experiments involving blanket variables. Provisions should also be made for easily altering the size of the core. This can probably be accomplished by providing a zone around the core which will permit flexibility of loading in this area. This zone should permit loading of core, reflector, or blanket materials interchangeably. The lattice spacing or pitch in the core and "intermediate zone" should permit loading different shaped elements. It appears possible to achieve interchanging cores consisting of fuel element cross sections which are rectangular, hexagonal or round.

Successful achievement of this approach should permit:

1. Variations in size of core and blanket.
2. Variations in configuration of fuel and blanket elements.
3. Variations in size and configuration of reflector, (if any).
4. Variations in mode of control.
5. Evaluation of variables, such as blanket moderation, blanket density, reflector materials, etc.

This degree of flexibility will impose some limitations which may prove to be incompatible with the over-all objectives and, therefore, dictate some compromises. A few of these can be recognized almost immediately, and appear to be as follows:

1. The core structure must be extremely simple--no lattice structure in the core proper or above.
2. Versatile unloading techniques will be required, and can probably only be achieved at the cost of speed and elegance.
3. The variety of reactor configurations may require some variation in control and safety methods, and these will probably require operation from the top only, limiting the types of unit which can be utilized.
4. It will probably be necessary to conduct all experiments from the top, imposing some limitations on the experimental equipment. The area outside the reactor vessel will have little experimental value.
5. Flexibility of loading will require flexible cooling of the various elements. This will probably require that the necessary orificing be accomplished in the element, rather than incorporating permanent orifices in the internal structure.

Reactor Vessel

The reactor vessel may be visualized as a large tank, shielded all around, and with appropriate inlet and outlet connections for the coolant system. The top of the vessel must contain a shielded section providing access to the interior of the vessel. This will probably take the shape of a large plug, and the desired flexibility of access will probably require the use of eccentric rotating plugs to permit flexible coverage of the entire vessel area. In addition, provisions must be made for multiple access to the interior for control and safety units, and miscellaneous experimental purposes. A certain

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degree of flexibility in location of these access ports will also be desired here, and can possibly be achieved by the use of several eccentrically located rotating plugs, inside a large rotating plug.

It would be desirable (perhaps necessary) to provide a secondary access port to the vessel, which can be flooded with the coolant. This would be a desirable facility for demonstrating the feasibility of fast unloading techniques, or other experimental applications involving severe cooling considerations.

Heat Removal System

It is generally agreed that the experimental reactor should have an intermediate coolant circuit in the heat removal system for flexibility and safety. The use of a secondary circuit of this type will permit the incorporation of experimental components in this circuit with less difficulty than could be accomplished in the radioactive primary circuit.

It can be shown that there are many advantages in a system containing a Na primary system, and a NaK secondary system, based on the excellent heat transfer properties of Na, and the low freezing temperature of NaK. To provide experimental flexibility in these coolant circuits, spare capped piping connections should be provided in the circuits. In the primary circuit these "dead" lines should lead into shielded cells, or into areas that can be shielded at a later date.

It would appear feasible to follow the original concept of providing a single primary circuit of large capacity (approximately 10,000 gpm) to gain experience with components which would have direct application in a full scale plant. It will probably be necessary to include a second stand-by circuit, or at least make the necessary provisions that such a system could be added at a later date. Provisions should also be made for taking off auxiliary circuits from the main system to accommodate experimental equipment for such purposes as analysis and purification of the coolant. This will require judicious planning, for all experimental equipment added to the primary circuit will require extensive shielding.

The secondary, or intermediate, system should be capable of considerable experimental modification, for it should be relatively simple to incorporate experimental components. It is also in the secondary circuit that flexibility of heat removal is required to provide flexibility in the operating temperature of the reactor.

Provisions must be made to accommodate contamination of the primary system. Eventually fuel elements will be incorporated in the core which countenance the probability of introducing long-lived fission product activity into the coolant. Provisions should be made for cleaning up such a system, and if necessary, permanently isolating part of a circuit from the reactor system, and switching the operation to auxiliary equipment. A contaminated coolant circuit must not be a prelude to the reactor's "swan song."

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-4-

Safety and Reliability

As is customary, a "fail safe" philosophy must prevail throughout the plant system. Components, such as controls, valves, instruments, must meet this requirement. Emergency cooling of the reactor after shutdown is always of vital importance, and provisions should be made here for the evaluation of various methods of accomplishing this. This is particularly important in reactors with a high power density, where cooling under normal operation is accomplished at high flow rates. The cooling requirements at shutdown are severe, and loss of normal pumping power must be assumed. Under these conditions complete reliability of heat removal is required. The Prototype might conceivably incorporate more than one method of accomplishing this to establish the reliability of various methods, and establish feasibility for full scale plant operation. The following are some of the methods which might be considered.

The intermediate heat exchanger might be made an unfailing "heat sink", and then utilize thermal convection in the primary circuit to transfer the heat from the reactor to the heat exchanger. Reliable heat removal from the heat exchanger could be accomplished by a gravity supply tank in the secondary system of large capacity. Another possibility would be a natural convection boiler in the secondary loop dumping some steam, and utilizing some steam to drive a steam turbine which in turn drives a homo-polar generator for an electromagnetic pump, which assists a natural convection circuit in the secondary. The pump could also "float" on storage batteries. There are undoubtedly many reliable methods for removing heat from the heat exchanger which should be investigated.

The above described approach relies on the primary circuit as a link in the emergency cooling cycle. It would certainly be advantageous to have a reliable component in the primary circuit for this purpose, and an emergnecy thermal convection loop in this circuit should be evaluated.

Methods of achieving reliability of pumps and other components, and divorcing their independence on the main power supply must also be investigated.

Since this is an experimental unit, reliability of continuous operation will be sacrificed to the extent that failure of equipment will be repaired rather than replaced by expensive idle stand-by equipment. The emphasis will be on serviceability and accessibility, and stand-by equipment will be held to a minimum.

The cooling system must not permit any rapid temperature changes in the reactor, particularly any rapid decrease in temperature. This can best be achieved by providing large heat capacity in the cooling system. Because it is desirable to keep the primary system as small and compact as practicable due to shielding considerations, the heat capacity probably can best be obtained in the secondary system.

LJK:jch

L. J. Koch

L. J. Koch

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-5-

To: W. H. Zinn

Dist: W. H. Zinn (4)
H. Etherington (1221)
A. Amorosi
R. Avery
A. H. Barnes
F. G. Foote
H. L. Hull
E. Hutter
A. S. Jameson
N. Hilberry
L. L. Kintner
L. J. Koch
S. Lawroski
M. Levenson
H. V. Lichtenberger
D. Okrent
J. F. Schumar
M. C. Shaw
F. A. Smith
K. F. Smith
B. I. Spinrad
C. R. Sutton
F. W. Thalgott
M. Treshow
L. A. Turner
S. Untermyer
J. M. West
RFD File 105
Reading File

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ATTACHMENT 5

MEMORANDUM: W. H. ZINN TO DISTRIBUTION, MARCH 9, 1953

RESTRICTED DATA

THIS DOCUMENT CONTAINS RESTRICTED DATA AS DEFINED IN THE ATOMIC ENERGY ACT OF 1946 ITS TRANSMITTAL OR DISCLOSURE OF ITS CONTENTS IN ANY MANNER TO AN UNAUTHORIZED PERSON IS PROHIBITED.

SECRET

Argonne National Laboratory

P.O. BOX 299
LEMONT, ILLINOIS

WHEN SEPARATED FROM ENCLOSURES, HANDLE THIS DOCUMENT AS ~~Unclassified~~

102

March 9, 1953

~~10~~

To: Distribution

From: W. H. Zinn, Laboratory Director

You are provided herewith with a copy of ANL-WHZ-353. This document gives some general considerations on the design of a FPBR and, beginning on page 9, sets forth in general terms the work program of the Laboratory. Further, on page 21, a table of manpower distribution is given.

Generally speaking, the information in ANL-WHZ-353 is sufficient for transmittal to the Atomic Energy Commission. For the proper coordination of the work in the Laboratory however, a general statement such as this is practically useless. Therefore, the work program has been subdivided into much more concrete topics. This has been done in a memo by L. J. Koch to me, dated March 4, 1953. The particular assignment of responsibility and the particular words in which the responsibility has been stated are not at all important. What you are requested to do by the present memorandum is to identify work you are doing for the PBR program in terms of the schedules in the March 4 memorandum. What you are hereby requested to do is the following:

1. Pick out those topics on which actual work is being done. If a suitable subheading does not exist, invent one.

2. Name the persons now assigned to the job. We should avoid subdividing people into many pieces.

3. Indicate the manpower assignment according to the distribution table on page 21 of ANL-WHZ-353. For fiscal 1953 this should include individuals actually working on the job or who may be expected to go to work before July. For fiscal 1954 it is the planned distribution.

Once this information has been provided and assembled, changes in assignments will be picked up as a result of the regular monthly distribution of effort. It is not expected that you will have to resubmit this information frequently. It is called for now so that the Laboratory can have a better understanding of the actual work going on in the project.

W. H. Zinn

W. H. Zinn

WHZ:fma
Enc. ANL-WHZ-353
Distribution: F. G. Foote
H. Etherington (2) w/o enc.
H. L. Hull
S. Lawroski
H. V. Lichtenberger
W. M. Manning
L. A. Turner
Reading file w/enc.
File

Mr. Etherington: Since copies of ANL-WHZ-353 have already been issued to L. Koch, A. Barnes, A. Amorosi and you, a copy is not being enclosed with this memo.
WHZ

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SECURITY INFORMATION

ATTACHMENT 6

MEMORANDUM: L. J. KOCH TO W. H. ZINN, MAY 28, 1953 (ANL-LJK-4)

SEC ... ON

PBR-10
W-A-30

101

CLASSIFICATION CANCELLED
BY AUTHORITY OF THE U.S., A.E.C., C.O.O.
BY Hoylande D. Young
DIRECTOR OF TECHNICAL INFORMATION
ARGONNE NATIONAL LABORATORY
DATE Feb. 10, 1958 R. J. Klrawski

ARGONNE NATIONAL LABORATORY
Lemont, Illinois
May 28, 1953

ANL-LJK-4
THIS DOCUMENT CONSISTS OF 7 PAGES.
NO. 8 OF 14 COPIES, SERIES A

To: W. H. Zinn — Laboratory Director

From: L. J. Koch — PBR Coordinator

Subject: PBR Program

Attached is a program outline covering the PBR program and supporting programs. This is a compilation of the Division Programs which were assembled in response to your memorandum of March 9 requesting specific manpower assignments to this work.

Coherence of activities was the motivating factor in preparing this outline, without primary concern for correlation with the usual budgetary scheme. The appropriate budget activity is indicated for each subdivision, and a customary manpower distribution table has been assembled based on this program outline. This summary appears at the end, together with a listing of the major supporting programs.

The listing of persons has been limited essentially to those devoting a minimum of 50% of their time to this work. The persons listed with an asterisk are contributing less than 100% of their time to this program.

The loaned employees from California Research and Development Company are listed in the most appropriate category describing their work. Some compromise was necessary in this respect, for a few of these men are working on programs somewhat removed from the PBR program.

L. J. Koch

Encl.
LJK:jch
Distribution:
1A. W. H. Zinn
2A. A. Amorosi
3A. A. H. Barnes
4A. F. G. Foote
5A. J. R. Gilbreath
6A. N. Hilberry
7A. H. L. Hull
8A. L. J. Koch
9A. S. Lawroski
10A. H. V. Lichtenberger
11A. W. M. Manning
12A. L. A. Turner
13A. E. C. Weber
14A. Reading File

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ANL-LJK-4

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Page 2

PBR PROGRAM

I. Physics

A. Theoretical and Analysis

1. Reactor Calculations - PBR, EBR, Critical Experiment
RE Nuclear - Avery, Greenspan, Krasner, Okrent, (Drummond, CR&D)
(4110-23)

2. Reactor and Coolant System Shielding Calculations
RE Nuclear - Grotenhuis (4110-71)

3. Exponential Experiment
PHY - Livingood,* Hummel,* Spinrad* (4110-21)
RE Mechanical - Brittan (4110-11)

4. Theory and Computing
PHY - Spinrad,* Hummel,* Rosenzweig,* Ferentz,* Dershem,* Feurzeig,*
McDowell* (4110-21)

B. Experimental Physics

1. Exponential Experiment
PHY - Livingood,* Lennox,* Martens, Beyer, Burgy, Nobles, Helfridge*
(4110-22)

2. Experiments at EBR

a. Activations, conversion, shielding, danger coefficients
RE Nuclear - Redman,* Curtis, Kaufman, (Farnham CR&D) (4110-23)

b. Neutron spectrum measurements
PHY - Ringo, Krohn, Huddleston (4110-22)

c. Alpha and (n, 2n) measurements on 233, 235, 238, and Pu-239
CHM - Fields* (5361-05)

3. Van de Graaff (Transport Cross-Sections, etc.)
PHY - Langsdorf, Mooring, Weeks,* Wallace,* Reardon,* Hibdon
(5261-01)

4. Isotopic Analysis
PHY - Ingraham* (5211-01)

5. Critical Experiment
EBR Project - Lichtenberger,* Novick,* Pettit,* Cerutti,* Cameron,*
McGinnis,* Whitham,* Haroldsen*

II. Reactor Materials

A. Fuel and Fertile Materials--Feasibility and Fabrication

1. Uranium Metal and Alloys - (Alloying, Workability, etc.)
MET - Yaggee, Mackerey, Dunworth (4110-32)

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2. Pu, Pu-U Alloys, and Ternary Alloys
   MET - Kelman, Wilkinson, Shuck (4110-32)

3. Liquid Fuels, and Slurries of U-Pu or their oxides in Na, NaK, Lead, etc.
   CHM - Abraham, Flotow, Katz,* Hyman* (5361-07)

4. Particulated Fuels - Balls, Powders, Pins
   RE Materials - Currier (4110-31)
   MET - Shuck, Yaggee (4110-32)

5. UF_6 Blanket Material (for continuous processing)
   CHM - Katz, Hyman (5361-03)

B. Fuel and Blanket Elements--Feasibility and Fabrication

1. Jacketing and Cladding Materials
   MET - Dunworth, Noland, Walker (4110-42)

2. Jacketing and Cladding Methods
   MET - Macherey, Noland, Walker, (Olson, Bean, Stone, Lawless CR&D)
   (4110-42)

3. Fuel Element Fabrication Techniques and Methods
   RE Materials - Monaweck, Dwight (4110-31)

C. Testing and Evaluation

1. Thermal Stability

   a. Thermal cycling of materials (basic)
      MET - Zegler (4110-32)

   b. Thermal cycling of fuel assemblies and components
      RE Materials - Fagan (4110-31)

   c. Thermal stability of particulated fuels - balls, pins, etc.
      RE Materials - Currier (4110-31)

2. Irradiation Stability

   a. Fissionable, fertile, and structural materials in EBR
      MET - Murphy (4110-43)

   b. High burnup and high temperature in MTR
      MET - Kittel* (4110-33)

   c. High burnup pin cushion experiments in CP-3'
      MET - Paine* (4110-33)

   d. Irradiation of fuel elements, prototypes, and components
      RE Materials - Monson, Stella, Matousek (4110-31)

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3. Physical Properties

   a. Irradiation effects on physical properties
      MET - Murphy,* Kittel,* Paine* (4110-33)

   b. Cladding and structural materials at elevated temperatures
      MET - K. Smith (4110-42)

   c. Physical properties plutonium and plutonium alloys
      MET - Kelman, Wilkinson (4110-32)

   d. Physical properties of particulated fuel elements
      RE Materials - Currier (4110-31)
      RE Heat Eng. - R. Rohde (4110-52)

4. Supporting Studies - Analysis, Basic Research

   a. Chemical analyses of plutonium samples
      CHM - Buchanan (5311-15)

   b. Oxygen and carbon analyses of plutonium
      CHM - Holt (5311-15)

   c. Fission product and plutonium analyses, EBR and experimental irradiations
      CEN - Vogel,* Krause,* Turk, Crouthamel,* Seefeldt (4110-24)

D. Processing of Reactor Materials

   1. Modification of Existing Processes to Accommodate Zirconium Alloys
      CEN - Levenson,* Vogel,* Seefeldt, Turk (4110-24)

   2. High Temperature Processes
      CEN - K. Rohde, Chellew, Glassner, Hampson, Feder,* (Jost CR&D) (4561-07)

   3. Chemical Recovery Processes for Irradiated Massive Plutonium
      CEN - Vogler, Shor, Mayer, Schraidt,* Farmakes,* (Volatility--Ludewig CR&D, Solvent Ext.--Christiansen CR&D) (4110-81)

   4. Metallurgical Separations - Pyrorefining - Electrolysis
      MET - Blumenthal, Marzano (5411-01)

III. Reactor Coolants (Na and NaK)

A. Composition and Purity

   1. Oxide Determination and Control
      RE Materials - Dzombak (4110-51)

   2. Fission Product Transfer In and Removal From Na Systems
      RE Materials - Sivetz (4110-51)

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3. Removal of Corrosion Impurities from Na
RE Materials - Scheibelhut (4110-51)

B. Corrosion and Mass Transfer

1. Corrosion of Uranium and Uranium Alloys - Static and Dynamic
RE Materials - ~~McGuire~~ (4110-31)

2. Corrosion at Elevated Temperatures
RE Materials - ~~Dzombek~~ (4110-41)

3. Mass Transfer at Elevated Temperatures
RE Materials - Scheibelhut (4110-41)

C. Special Problems

1. Self Welding, Galling of Materials in Na
RE Materials - Neisz (4110-41)

2. Cleaning Extensive Systems of Na
RE Materials - F. Smith, (Harris CR&D) (4110-51)

3. Cleaning of Active Fuel Elements and Disposal of Active Na
RE Materials - Sivetz (4110-51)

IV. Heat Engineering--Heat Transfer and Power Cycle

A. Theoretical - Analysis and Design

1. Coolant System Conceptual Design and Analysis
RE Heat Eng. - Kintner, Simmons (4110-52)

2. Steam Cycle Conceptual Design and Analysis
RE Heat Eng. - Simmons, Hastings (4110-52)

3. Shutdown and Emergency Cooling Analysis
RE Heat Eng. - Hastings (4110-52)

4. Stress Calculations - Fuel Elements and System Components
RE Heat Eng. - Flinn (4110-52)

5. System Components - Heat Exchangers, etc., Design and Analysis
RE Heat Eng. - Kintner (4110-52)

B. Experimental Heat Engineering

1. Heat Transfer Measurements - Shapes, Wetting, Burnout, etc.
RE Heat Eng. - Kuczen, R. Rohde (4110-52)

2. Over-all Heat Transfer Coefficients - Heat Exchangers, etc.
RE Heat Eng. - Kuczen (4110-52)

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3. Steam Leaks into NaK Under Simulated Steam Generator Conditions
RE Heat Eng. - Kuczen (4110-51)

4. Pressure Drops - Fuel Assemblies, Heat Exchangers, Boilers, etc.
RE Heat Eng. - R. Rohde (4110-51)

V. Mechanical Engineering and Design

A. Analysis and Design

1. Reactor and Plant Arrangement
RE Mechanical - Winkleblack, Hutter, (Steckler CR&D) (4110-11)

2. Fuel and Blanket Element Design
RE Mechanical - Hutter, (Wellman CR&D) (4110-31)

3. Design of Experimental Mockup of Primary Cooling System
RE Mechanical - Jamrog, Soppet, Dickson (4110-11)

4. Design of Reactor Loading, Unloading, and Supporting Facilities
RE Mechanical - Winkleblack, Nicoll (4110-11)
RCD - Goertz* (4110-35)

5. General Instrumentation Layout
RE Mechanical - ~~Spalding~~ (4110-11)

6. Control Elements and Control System Components
RE Mechanical - Winkleblack, Jamrog, ~~Spalding~~ (4110-61)
RCD - Harrer* (Snider CR&D) (4110-62)

B. Experimental and Test

1. Mockup of Loading and Unloading Scheme
RE Mechanical - Dickson, Nicoll (4110-11)

2. Mockup of Complete Primary - Reactor System
RE Mechanical - Dickson, Soppet, Jamrog (4110-11)

3. Construction and Testing of Large EM Pump
RE Mechanical - Soppet (4110-11)

4. Large Valve, and Other Piping System Component Tests
RE Mechanical - Dickson (4110-11)

5. Control System Component Tests
RE Mechanical - Jamrog, Spalding (4110-61)

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Manpower Distribution

SUMMARY

RE 36.5
PHY 11.5
CEN 7.0
MET 14.0
R.C. 1.0
EBR 4.0
74.0

ANL-NH-121 (54)

PBR Program:

Budget Activity	Division	FY-1954	
4110.11-Over-all Design & Evaluation	RE	7.0	6.0
4110.21-Theoretical Physics	PHY	3.5	3.0
.22-Experimental Physics	PHY	8.0	5.0
.23-Experimental Physics & Analysis	RE	6.5	6.0
.24-EBR Analysis	CEN	3.0	4.0
.25-EBR Physics Experiments	EBR	1.0	1.0
4110.31-Fuel & Producer Element, Design & Testing	RE	8.5	8.0
.32-Fuel & Producer Element, Metallurgy	MET	7.0	6.3
.33-Fuel & Producer Element, Irradiation Effects	MET	2.0	3.2
.34-EBR Fuel Element Experiment	EBR	2.0	2.0
.35-Remote Handling Studies	RC	0.5	
4110.41-Structural Materials Tests	RE	2.5	7.0
.42-Structural Materials Metallurgy	MET	4.0	3.3
.43-Structural Materials Irradiation Effects	MET	1.0	3.2
4110.51-Coolant Studies	RE	3.5	4.5
.52-Heat Transfer Studies	RE	5.0	7.0
.53-EBR Tests on Liquid Metals	EBR	1.0	1.0
4110.61-Instrumentation & Control	RE	2.5	9.0
.62-Remote Control Instrumentation	RC	0.5	1.0
4110.71-Shielding Development	RE	1.0	2.5
4110.81-Chemical Processing	CEN	4.0	6.0
		74.0	89.0

RE 26.5
P. 8.5
CEN 6.0
EBR 5.0
R.C. 0.5
MET 3.5 – (4.0)
50.0

RE - 50.0
PHY 8.0
CEN 10.0
Met 16.0
R.C. 1.0
EBR. 4.0
89.0

Supporting Programs:

Budget Activity	Division
4520.02-Development of Methods for Reactor Calculations	PHY
4581.07-High Temperature Processes	CEN
4711.01-EBR Operations	EBR
5211.01-Neutron Physics & Nuclear Spectroscopy	PHY
5261.01-Determination of Nuclear Constants	PHY
5311.15-Physical Chemistry & Physical Measurements	CHM
5361.03-Laboratory Studies on Separation Processes	CHM
.05-Properties of Nuclei Relating To Reactor Operations	CHM
.07-Properties of Reactor Materials	CHM
5411.01-Basic Metallurgy of Project Materials	MET

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ATTACHMENT 7

MEMORANDUM: L. J. KOCH TO N. HILBERRY, SEPTEMBER 2, 1953 (ANL-LJK-6)

CLASSIFICATION CANCELLED
BY AUTHORITY OF THE U.S., A.E.C., C.O.O.
BY Haylande D. Young
DIRECTOR OF TECHNICAL INFORMATION
ARGONNE NATIONAL LABORATORY
DATE Feb. 10, 1958

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Security Information

Argonne National Laboratory

ANL-LJK-6
This Document consists
of 9 pages. No. 11 of
19 copies. Series A.

102

September 2, 1953

To: N. Hilberry

From: L. J. Koch — Deputy Laboratory Director

PBR Coordinator

Subject: Power Breeder Reactor (PBR) Program.

In answer to your request, the following summary of the present PBR program has been prepared. It includes a description of the major development work now planned, and a time schedule of the required experimental facilities.

Purpose

start

The ultimate purpose of this program is to develop a Fast Breeder Power Reactor which meets the requirements of both technical and economic feasibility. It is intended that economic feasibility means the sale of central station power, competitive with existing power sources, with by-products having only a commercial market.

The immediate purpose of the program is to develop, construct, and operate an Experimental Fast Power Reactor which will demonstrate the technical feasibility of these reactor plants.

Summary

Results obtained to date in the fast reactor program indicate that a sizable development program is required, and that present facilities are inadequate to successfully solve the technological problems involved. These results also indicate the course which this program must take.

The physics of fast reactors must be extended. There is much to be learned regarding the basic nuclear reactions in a fast neutron spectrum. The techniques of theoretical analysis of fast systems must be improved, but of even greater importance is the need for additional basic measurements of the nuclear phenomena. An accelerated experimental program is indicated. This program requires the use of specialized facilities, and materials about which little is now known.

Plutonium is the favored fuel. This has been suspected for some time, but recent measurements have indicated that the superiority of plutonium over U-235 is greater than originally suspected. The metallurgy of plutonium and plutonium alloys, as applied to reactors, must be advanced tremendously. Facilities will be required to undertake this work.

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Security Information

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Fuel reprocessing must be simplified. The economic feasibility of the fast reactor cycle cannot be achieved without the technical development of a simplified low cost reprocessing operation.

Fuel fabrication must be simplified. Here again the economic feasibility is dependent upon the technical development of new, low-cost fabrication techniques. The reprocessing cycle will probably impose the additional requirement of fabricating the fuel from only partially decontaminated materials.

Large scale system components must be developed. This will require more than a simple extrapolation or scale-up of existing equipment. Particular effort is required on large pumps, valves, and heat exchangers satisfactory for use in a large plant suitable to central station power applications.

Irradiation damage resistant materials must be developed. This problem is not peculiar to the fast reactor, for it is common to most of the reactor types being considered for central station power applications, however, even though some of the basic work being done in this field will be applicable or of assistance to the fast reactor program, a sizable experimental program is required specifically for this program. The environment of a fast reactor, and the desired composition of this reactor type is quite different.

Operational experience is required. More experience, and thus more confidence, is needed as far as fast reactors are concerned. The EBR has been invaluable in this respect, but operational experience with a machine whose "modus operandi" is more like that to be encountered in central station power applications is needed.

Additional experimental facilities are required. The development program involved requires experimental facilities in addition to those now available at ANL. The nature of this work requires integrated experimental facilities which will permit the correlation of a flexible, dynamic, development program. The additional facilities required are: Fast Exponential Assemblies, Zero Power Critical Assemblies, Large Scale Liquid Metal Component Test Facilities, Plutonium Fabrication and Processing Facilities.

Program

The following is a detailed description of the program which is under way, or planned, as a part of the Fast Power Breeder Reactor Development.

Physics

The EBR is the most useful experimental facility now available for the fast reactor program. It is, however, a highly enriched, compact assembly with a configuration and composition quite different from that required for the PBR. Its usefulness should not be minimized, however, for much information and experience has been gained in analyzing the performance characteristics of the EBR. The conversion ratio has been

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measured on a U-235, uranium, plutonium cycle. Alpha, the ratio of capture to fission, has been measured for U-235 and for plutonium. Important operating characteristics such as the reactor temperature coefficient have been measured and correlated with theoretical predictions. The difficulty associated with the above described results, concerns the interpretation of these measured values, and extension to other fast reactor designs. Probably the most vital link is a knowledge of the neutron spectrum in the EBR. This is an extremely difficult measurement, but the effort which is being expended to make this determination is indicative of its importance to the program. If these measurements are successful, it will be possible to predict the performance of other fast assemblies to a higher degree of accuracy than now obtains.

To obtain additional information regarding the characteristics of large dilute fast assemblies, an exponential experiment is now being constructed. This is admittedly an intermediate step, and will not provide all the necessary information required in the design of a full scale fast reactor. This subcritical experiment, however, can be justified on the basis that it is far simpler to construct and operate than a critical experiment; it requires a smaller inventory of critical materials; and it permits utilization of much existing equipment.

It is expected that the exponential experiment will permit obtaining the following properties of dilute fast assemblies:

1. Bare critical mass as a function of dilution and composition.

2. Reflector savings of various reflector materials.

3. Effects on buckling of various control schemes, core composition, and core geometry.

4. Neutron Spectra by activation of threshold detectors and possibly by direct measurement of beams.

5. Cross-section data of a semiquantitative nature. (These will be measurements of gross effects due to the incorporation of various materials in the assemblies.)

Unfortunately it will not be possible to obtain some of the basic data such as: microscopic cross sections, conversion ratios, alpha, or detailed flux behavior suitable for reactor analysis. (An excellent example of the program correlation required is demonstrated here. If the spectrum and corresponding alpha is determined in the various parts of the EBR, then alpha for the given spectrum can be applied to this experiment. There is no known method for determining alpha in a low power exponential or critical experiment.)

It is expected that the results obtained from the exponential experiments will permit the design of a better core configuration then is now possible, and permit the optimization of these configurations. It is not expected, however, that such analyses can be carried out with

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a sufficient degree of accuracy to permit direct detailed design of the reactor arrangement. A zero power fast critical assembly will be required to verify, and probably modify, these predictions. Plans are already under way for the construction of a critical assembly facility to fill this obvious gap in our nuclear know-how. It is planned to make this critical assembly facility as flexible as possible, to permit the final analysis of the many variables involved in the reactor arrangement, and to permit the detailed design of the actual reactor components.

Reactor Fuel

Plutonium ultimately will be the desired fuel for the PBR. Unfortunately very little is known about the metallurgy of plutonium and plutonium alloys as applied to the detailed design of reactors. Present knowledge and techniques, as applied to plutonium, have been developed for a purpose quite different than required for reactor applications. Experience to date has shown that it is most difficult even to obtain small experimental samples for reactor work from present facilities.

Techniques for the fabrication, alloying, and processing of plutonium must be developed for reactor applications. Of particular interest to the fast breeder program is the basic metallurgy of uranium-plutonium alloys and uranium-plutonium-ternary alloys, assuming that suitable stabilizing elements may be required. A large scale development program is indicated for these materials, in order that such basic information as: irradiation stability, corrosion, fabricability, and basic physical properties of these materials may be determined.

The experience gained to date with uranium and uranium alloys is of course applicable to this program. The operating characteristics of reactors can be obtained from uranium fueled machines. Some extrapolation to the operating characteristics of plutonium fueled machines will be possible, particularly when the basic properties of the plutonium alloys of interest have been determined. It is also very likely that enriched uranium will probably serve as a "interim fuel" in the PBR program, with a conversion to plutonium when the technology of plutonium has advanced sufficiently.

Processing

It has become somewhat axiomatic that the success of the PBR is dependent upon the economic feasibility of the reprocessing cycle. It also has been generally accepted that the reprocessing philosophy must be considerably different than that required at present. Simplified processes for irradiated uranium are being investigated and evaluated. High temperature volatility processes are being developed. Studies will be made of decontamination obtained by melting in the presence of salts. Electrorefining processes are also being investigated. The most promising methods must be extrapolated to the recovery of irradiated massive plutonium. This will be a hazardous operation, and will require facilities which are unavailable anywhere at the present time.

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Fabrication

The most influential criterion established for the fabrication process is that the fuel must be processed by remote controlled methods, and only partially decontaminated material will be available to the fabricator. Fabrication processes are being investigated to determine methods which are applicable to this type of operation. Some compromise is clearly indicated here, between the desires of the fabricator and the reactor designer, but to be successful, the entire PBR cycle must meet the objectives of technical and economic feasibility.

Components

One of the more important engineering problems involves the development of large scale system components. A large amount of work has been done to perfect units for the applications required to date, and the reliability of these units has been demonstrated. Experience at KAPL has indicated that the extrapolation from component sizes required for the EBR coolant system, to the sizes required for the SIR, has created some formidable problems. Coolant system components, such as valves, pumps, and heat exchangers cannot be scaled up directly without considerable difficulty, and apparently involve engineering problems which are directly associated with the size of the equipment. Also, the direct scale-up of valves, for example, results in units of horrendous size and of questionable feasibility. New design concepts, and engineering tests of these concepts will be required. The feasibility of large scale components for 12 to 16 inch pipe size sodium systems must be demonstrated, as well as large pumps of the order of 10,000 gpm capacity.

Irradiation Damage

The PBR must be a high burn-up reactor. A burn up of at least 2 per cent (total atoms) probably must be achieved for a successful cycle. Experimental evidence has indicated that this is not an unreasonable objective with uranium alloys in a thermal reactor. It is also encouraging, that the limited data obtained thus far, indicate that some plutonium alloys may be more irradiation damage resistant than uranium alloys. Unfortunately these evaluations cannot be carried to completion in a fast neutron flux. Correlations are being made between irradiation damage in the EBR and thermal reactors, but the comparisons are at low burn-ups. The reliability of extrapolation to high burn up is somewhat questionable, but this problem cannot be solved (in a reasonable time) until a fast neutron irradiation facility with a flux of approximately 10^{16} (estimated PBR flux) is available. This exceeds the EBR by a factor of almost 100.

The PBR program will certainly benefit by the basic irradiation damage program being conducted on uranium alloys. The PBR program must essentially "go it alone" as far as the plutonium alloys are concerned, particularly of the composition of interest to this reactor.

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Operational Experience

There is no substitute for the experience and confidence which can be gained by the operation of an actual operating unit. The existence of the EBR could probably be justified on this basis alone. The EBR, however, is not operated as a central station power plant, nor was it intended to be. The EBR is an experimental unit with a secondary objective of demonstrating the engineering feasibility of some small scale components which are similar in nature to those required for a power plant.

It will be necessary to demonstrate a complete experimental operating unit which will serve essentially as a prototype version of such a full scale power plant. This experimental pilot system must be large enough to permit extrapolation of components to full scale unit without further development. It should include the facilities for the entire PBR cycle; the reactor, the processing cycle, and the fabrication facility. In addition it must be sufficiently flexible to permit modification of processes and reactor design, and thus serve as an experimental facility to the extent that future improvements and advancements in technology can also be demonstrated in the longer range program.

Long Range Studies

It has been generally agreed that a very bright future for the Fast Power Breeder is assured, if some of the presently undemonstrable ideas can be made to materialize. In this catagory are: liquid fuels in homogeneous or heterogeneous reactors, continuous reprocessing, very high temperatures, and simplified power cycles. It is apparent that some effort should be directed toward the successful accomplishment of these goals, for although the tasks are extremely difficult, the rewards are probably more than commensurate with the efforts required.

To maintain a healthy and vigorous immediate program, it appears advisable to keep these long range objectives separated from the immediate and better definable program. These long range problems should be attacked at a slower pace until such time that the direction of development can be established, and man power can be efficiently utilized toward the solution of the defined problems.

Facilities

The program outlined above cannot be conducted without specialized experimental facilities. It is equally obvious that this work must be correlated, and that this can only be accomplished efficiently if these facilities are integrated. For example, the correlation of core design, fuel processing, and fuel fabrication are so closely interrelated that it is vertually impossible to accomplish this difficult task unless there is day to day contact and interchange between the three groups involved in this phase of the work. This development will require the ultimate in flexibility of operation, and liaison of a type which can only be accomplished by people who are literally "rubbing elbows" with the people working on the other phases of the program.

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The experimental facilities required as a part of the PBR Program follow, with a realistic time schedule which indicates the interconnection between the phases of the program.

Fast Exponential Experiments

This facility is now being constructed in building 316. Results should be forthcoming in the second half of fiscal 54. This experiment utilizes much of the equipment of the ZPR-1 experiment and entails no major construction.

Fast Critical Assemblies

This facility is the "Follow-up" required to obtain the necessary reactor physics information, and is now in the planning stage. This unit will require the construction of a rather modest building which will provide a very flexible facility similar to our present building 316. It will be located near the EBR and will utilize some of the facilities at that installation.

Construction on this facility should be started during this fiscal year in order that the first experiments may be conducted in fiscal 55.

Plutonium Facility

If plutonium is to find application as a reactor fuel, it will require this facility to accomplish the necessary development. The importance of this experimental facility cannot be overemphasized, for the task of producing a usable reactor material from plutonium is formidable indeed. Processing and fabrication must go hand in hand as far as reactor applications are concerned, and essentially nothing has been done toward solving the problems of processing massive irradiated plutonium. It is unfortunate that nature endowed plutonium with such favorable nuclear properties, and such unfavorable biological properties, but the solutions to the latter problems must be found. This will be done only by a very concentrated effort, requiring the specialized facilities demanded by the product.

Construction on this facility should be started early in fiscal 55 in order that this work can be under way during fiscal 56.

Mock-up Test Facilities

This is to be an engineering facility to develop and demonstrate the technical feasibility of large scale components associated with the coolant system of a full scale reactor plant. As previously stated, the feasibility of these units must be demonstrated in full scale operating experiments prior to installation in a reactor plant where absolute reliability is demanded of them.

The pumps, valves, and other components can be developed and tested on a reduced scale in building 206, our liquid metal experimental facility.

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The extrapolated full scale models must be demonstrated in a full scale system. Building 206 cannot handle this equipment for it is an experimental facility for liquid metal experimental work. This work will require an addition to building 206, or a separate facility, for testing large liquid metal systems and components.

Another vital need for this facility is to provide a suitable place to mock-up the reactor installation, and demonstrate the loading and unloading techniques, and other handling operations associated with reactor operation.

Construction of this facility should begin early in fiscal 55 in order that it will be available before the end of fiscal 55. Many of the final designs of components for the experimental reactor will be dependent upon the results of tests conducted in this facility. In many cases, final engineering approval will be obtained here.

Experimental Fast Power Reactor

This is the objective of our immediate program. It is to be an experimental plant which will also serve as a Pilot Plant for the full scale Power Breeder Reactor. This will be a plant of approximately 50,000 KW power (heat) which is approximately 10 per cent of the size of a reasonable full scale PBR. This will be a fully integrated plant including the reactor and complete power cycle, the processing and fabrication facilities for supporting the operation, and associated equipment. It will be flexible in operation, both temperature and power level (and thus power density) may be varied. It will be flexible in construction, in order that variations in reactor geometries and compositions may be evaluated. Probably the most important feature will be that it will operate on uranium, or plutonium in many forms. This is the direction of greatest advancement for the Fast Power Breeder Program, and the Experimental Reactor will be capable of evaluating and demonstrating the feasibility of these advanced designs.

Construction of this facility should begin early in fiscal 56 (probably with uranium as first fuel charge). The unit can then be in operation in fiscal 57.

The schedule outlined above assumes a continuing man power effort at approximately the presently proposed level, with an increase required in fiscal 56. During this period all the experimental facilities will be in operation, and a very active experimental program will be in progress. In addition a very active design and procurement program will be in progress to meet the construction schedule on the Experimental Reactor Plant.

L. J. Koch

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Mr. Hilberry

September 2, 1953

Distribution

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ATTACHMENT 8

EXCERPTS FROM

"THE ENGINEERING DESIGN OF EBR-II, A PROTOTYPE FAST NEUTRON REACTOR POWER PLANT"

BY

A.H. BARNES
L.J. KOCH
H.O. MONSON
F.A. SMITH

AVAILABLE IN THE PROCEEDINGS OF THE
INTERNATIONAL CONFERENCE ON THE PEACEFUL USES OF ATOMIC ENERGY,
VOL. 3, "POWER REACTORS," (AUGUST 8–20):380–344, 1955.

The Argonne Experimental Breeder Reactor II (EBR-II) is one of the five prototype industrial power reactors selected by the United States Atomic Energy Commission for development and construction. The EBR-II is a plutonium-fueled, unmoderated, sodium-cooled reactor with a power rating of 60,000 kw (heat). The plant consists of the reactor and heat removal system, the steam-electric power plant, and an integral fuel reprocessing facility. The plant is now in the engineering design and development stage.…

GENERAL PLANT DESIGN

The EBR-II Plant is shown in Fig. 1. It is divided into four major systems which may be defined as follows:

1. The Primary System: the reactor and the primary sodium cooling system.
2. The Secondary System: the intermediate sodium heat transfer system.
3. The Steam System: the steam-electric system.
4. The Fuel Process System: the fuel recovery and fabrication facilities.

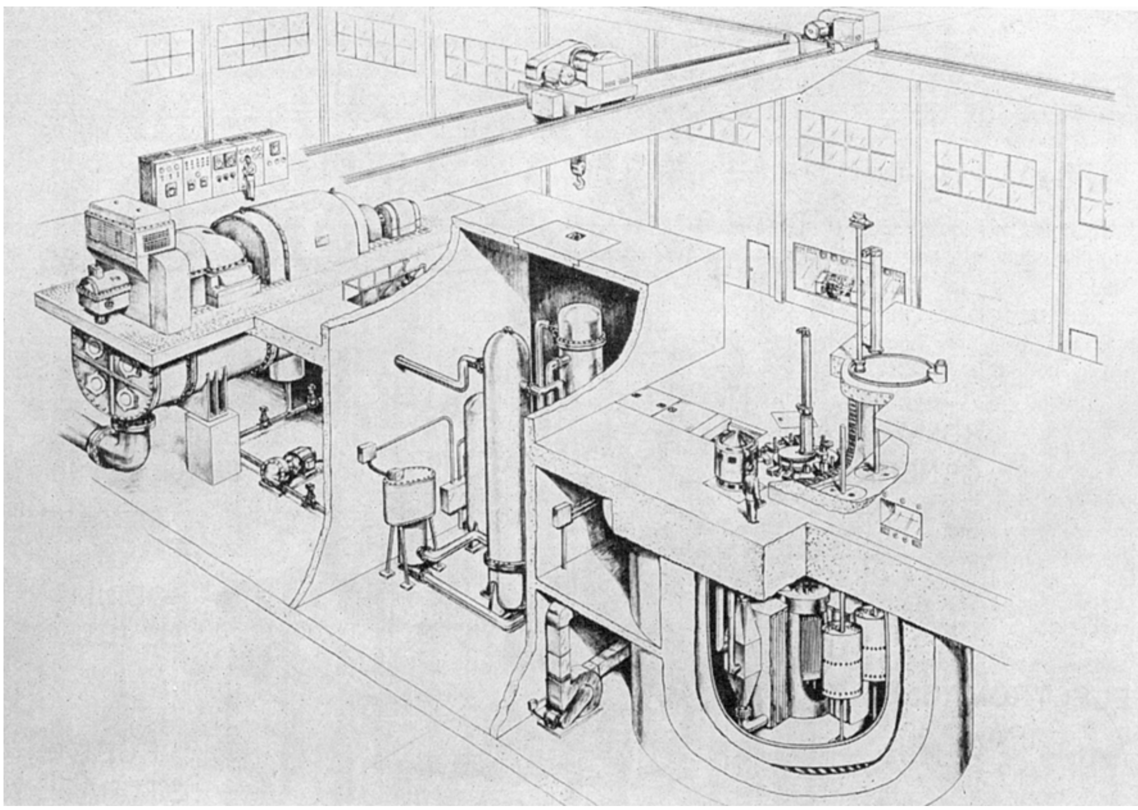

FIGURE 1. THE EBR-II PLANT.

The primary system is contained in a single vessel (referred to as the primary tank). All of the components in the primary system, including the reactor, the primary sodium pumps and piping, the heat exchanger, and the fuel transfer and storage system, are submerged in sodium as shown in Fig. 2.

The reactor consists of an enriched core of uranium-plutonium alloy, in the approximate shape of a hollow cylinder, surrounded on all sides by a uranium breeding blanket, as indicated in Fig. 3. The average core power density is approximately 1,000 kw/liter, and the average core heat flux is approximately 1×10^6 Btu/ft^2-hr. Reactor cooling is a critical problem, not only during operation, but also after shutdown.…

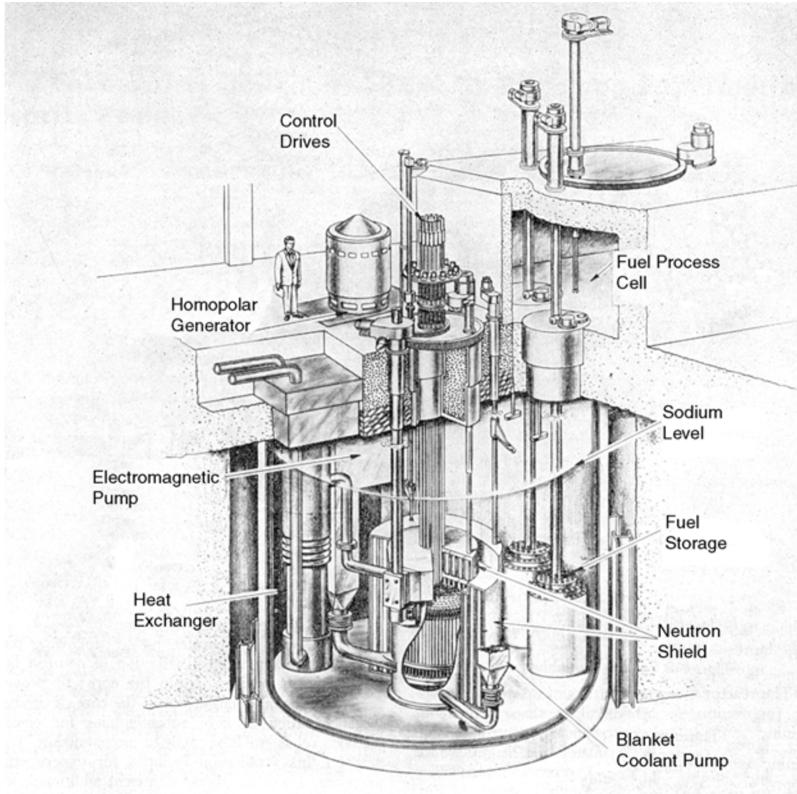

FIGURE 2. EBR-II PRIMARY SYSTEM.

REACTOR

The reactor is divided into four main zones, viz., Core, Central Blanket, Inner Blanket, and Outer Blanket, as indicated in Fig. 4. Each zone is comprised of a number of right hexagonal subassemblies containing the fuel or blanket elements. All subassemblies are of identical size. Their numerical distribution is as follows:

Central Blanket	=	7
Core	=	42
Control	=	12
Inner Blanket	=	66
Outer Blanket	=	510
Total		637

An annular core with a central blanket of uranium has been incorporated for the following purposes: to flatten radial distribution of neutron flux and power generation within the core; to provide for experimental enlargement of the core, if desired, by substitution of core subassemblies for central blanket subassemblies; and to accommodate high neutron flux irradiation facilities for experimental purposes. There are thought to exist additional, less direct advantages of a central blanket, as well.…

Because power densities within the central and inner blankets are similar in magnitude, identical subassemblies (both in composition and construction) are used in these zones.

A single subassembly size is employed throughout the entire reactor because: a maximum of flexibility in reactor configuration is obtained; the amount of parasitic reactor volume is reduced; and subassembly handling equipment and reactor tank structure are simplified.…

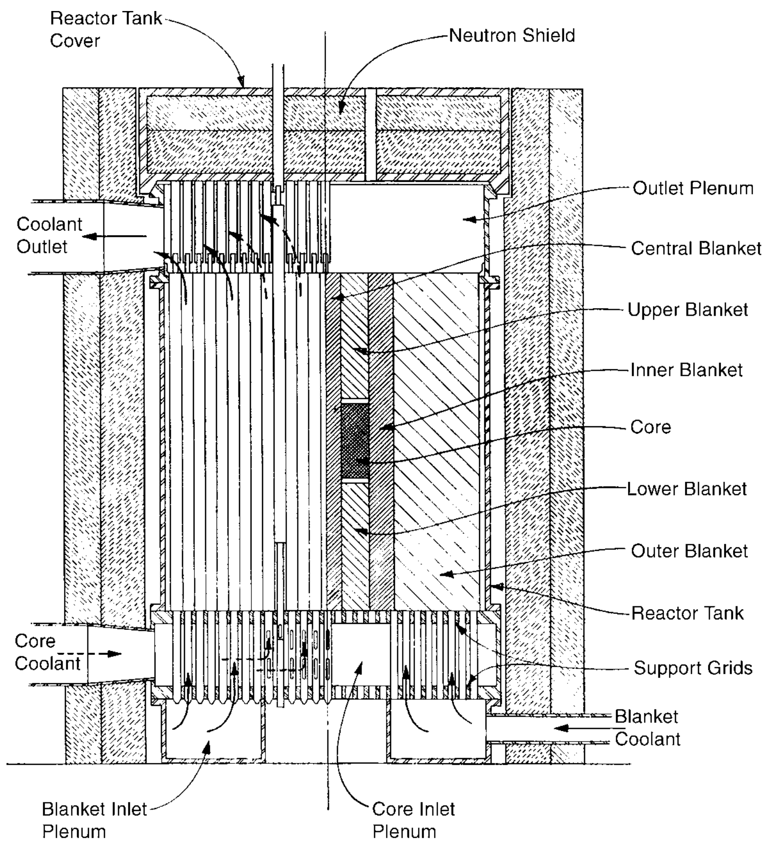

FIGURE 4. SCHEMATIC ELEVATION OF EBR-II REACTOR ASSEMBLY.

PRIMARY COOLANT SYSTEM

The primary coolant system includes the reactor, the intermediate heat exchanger and the circulating pumps. A significant feature of this design is the close-coupled arrangement of the components of the primary system in the primary tank in which they operate submerged in liquid sodium.…

Coolant is pumped directly from the bulk sodium in the primary tank to the two reactor inlet plenum chambers. After flowing upward through the reactor into the common top plenum chamber, it passes through the intermediate heat exchanger and then returns to the primary tank. The flow through the reactor is monitored by electromagnetic flow meters provided with insulating jackets to permit submerged operation.

The design requires that the pump be located within the primary tank and that it be capable of operating submerged in sodium without any form of auxiliary cooling other than that provided by the passage of the liquid being pumped. A d-c conduction type electromagnetic pump is employed, with sufficient capacity to handle the pumping requirement of the main coolant system. The pump, shown in Fig. 10, has a capacity of 10,000 gallons per minute at a head of 75 psi. The pump duct is 6 inches by 18 inches in cross section and carries the liquid at a velocity of 31 ft per second. The current is approximately 250,000 amperes required to drive the pump is supplied by a homopolar generator located near the top of the primary tank shield. The very large current requirement makes it necessary to place the pump and generator as closely together as possible. Consequently, the pump is located directly below the generator near the level of the sodium surface in the primary tank. The current is carried from the generator downward through the shield in sodium-filled ducts which attach to the pump housing. The ducts are partly filled with copper bars to provide adequate gamma shielding. The pump, generator, and drive motor are connected so that they can be individually removed should servicing become necessary.…

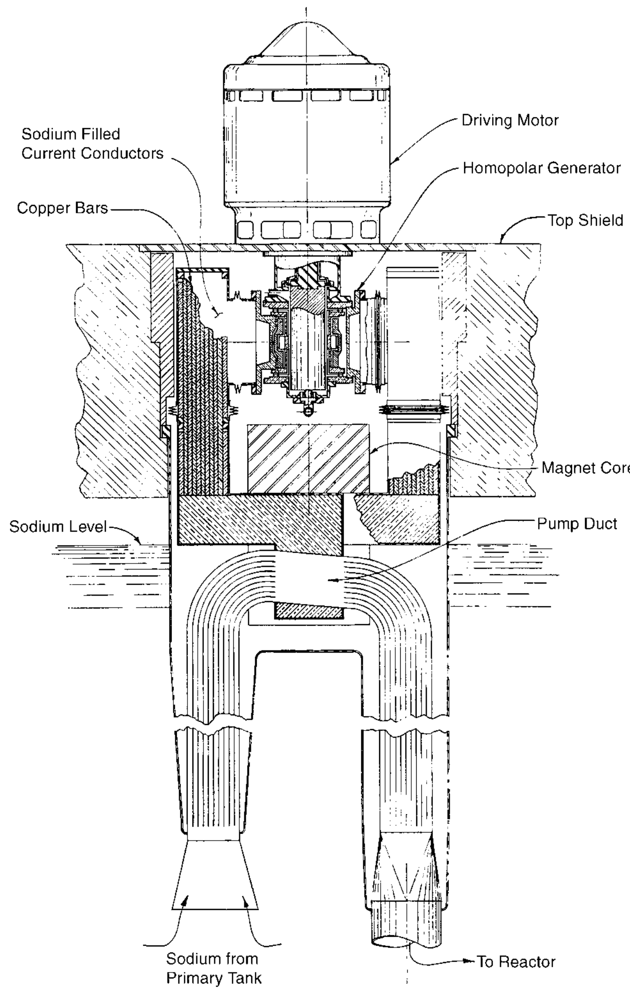

FIGURE 10. PRIMARY SYSTEM ELECTROMAGNETIC PUMP AND GENERATOR.

FUEL PROCESS SYSTEM

It is desirable in a fast power reactor to recycle the fuel as rapidly and as economically as possible to minimize the costs of fuel inventory and of fuel processing. Because of the high specific activity of the fuel from the reactor, a long "cooling time" is required if the fuel is processed by present aqueous methods, and a large economic penalty results because of the large fuel inventory required for cooling.

The EBR-II plant employs "pyrochemical" processing which involves high temperature slagging in the molten metal phase. The costly conversion steps required in aqueous systems are avoided, and the size and complexity of the equipment are reduced considerably. However, the process is not as efficient, and complete decontamination is not obtained. The results to date indicate that in excess of 90 per cent of the fission product activity can be removed by the process. This is an acceptable decontamination for fast reactor operation, because it is relatively insensitive to fission product poisoning. Sufficient residual fission product activity remains in the fuel, however, to require shielded remote control fabrication of the alloy. After processing, therefore, it is necessary to reconstitute and fabricate the material and assemble the fuel elements, completely by remote control methods. The fuel is fabricated into pins, loaded into the fuel element tubes, and assembled into the subassemblies. Obviously, as much preparatory work as possible is accomplished outside the fuel process cell. The structural materials are completely fabricated outside, and the remotely controlled assembly operation is limited to assembly of the fuel elements and the subassemblies.

Since the EBR-II is fueled with plutonium, semi-remote operation and complete containment would be required even if complete fission product decontamination was obtained. Therefore, the requirement of complete remote control operation due to residual fission product activity does not represent a large additional burden.

ATTACHMENT 9

MEMORANDUM: W. H. ZINN TO A. H. BARNES, JANUARY 4, 1956

Argonne National Laboratory
P.O. BOX 299
LEMONT, ILLINOIS

A. H. Barnes | Director, Reactor Engineering Div.
W. H. Zinn | Laboratory Director
January 4, 1956

EBR-II

Two matters related to the work of your division require decision by me. These are: (1) An architect-engineer for the EBR-II reactor power plant, and (2) the formal establishment of a project for the EBR-II reactor power plant.

These are not independent since the establishment of a formal project must be no later than coincident with the procurement of an architect-engineer. The main problem, therefore, on which a decision must be made is "Is the feasibility of the EBR-II reactor now sufficiently well established to justify the expenditure of sizable sums of money on an architect-engineer?"

For the purpose of resolving this question, your division has prepared a draft of an architect-engineer prospectus. With a few corrections and additions, the proposed prospectus is quite suitable for the purpose and, indeed, represents an excellent summary of the problems to be undertaken by the architect-engineer.

In order to help me in coming to a decision as to where we stand on the feasibility of this reactor, I would like to set up a series of meetings for the purpose of reviewing all of the vital points. I have in mind breaking it down into a list such as follows and restricting each meeting to a discussion of one or two of these points; in this way, the number of individuals from your division who will need to attend the meeting will be minimized. In order to broaden the area in which questions will be raised, I intend to appoint a Feasibility Evaluation Committee whose purpose it will be to advise me of its opinion in various areas in which it is expert. Clearly for this committee not to be too prejudiced, it cannot be made up of persons who work only on the EBR-II work.

I expect to hold meetings at 10:00 a.m. in the Building 10 Conference Room or in the Building 208 Conference Room on the following dates: January 16, 18, 20, 23, 25, 27, February 1, 3. Topics for these meetings are as follows:

January 16	Primary circuit layout and fuel handling
January 18	Control mechanism and reactivity coefficients
January 20	Size of core and blanket, heat transfer, and shielding of reactor, including biological shield
208 — January 23	Fuel element
January 25	Pumps and emergency and shutdown cooling
January 27	Fuel decontamination
300 PM — February 1	Remote fuel fabrication
February 3	Fuel cycle, economics, and holdup.

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2/17

FILE
DCV
JAN 4 — 1955
L. J. KOCH
REACTOR ENGINEERING DIV.
ACTION

January 4, 1956

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A. H. Barnes

If necessary, additional meetings will be held during this interval or it will be extended.

If you are agreeable to this scheme, the evaluation committee will be appointed at once and work will begin on January 16.

W. H. Zinn

WHZ:fma

cc: L. J. Koch
Reading file
File

ATTACHMENT 10

MEMORANDUM: W. H. ZINN TO L. J. KOCH, JANUARY 9, 1956

ARGONNE NATIONAL LABORATORY
Lemont, Illinois

January 9, 1956

FILE A-21
DCV
JAN 10 1955
L. J. KOCH
REACTOR ENGINEERING DIV.
ACTION

To: L. J. Koch

From: W. H. Zinn, Director

Subject: Feasibility Evaluation Committee for EBR-II

The purpose of this note is to ask you to serve on a Feasibility Evaluation Committee for EBR-II.

As you know, it is our procedure to hire an architect-engineer to prepare the detailed drawings for the construction of a reactor. This step is the first which commits funds in sizable quantity over and above our normal operating budget and, therefore, is to be taken only on the conviction that a satisfactory basic design has been evolved.

The EBR-II section of the Reactor Engineering Division has prepared a summary of the present design in the form of a prospectus of scope of work for an architect-engineer. The proposed prospectus gives a fine summary of the present state of the design. It does not, however, include evaluations of the basic performance of the reactor.

It is my intention to examine the design through a series of meetings beginning January 16 and taking place every other morning. I will act as chairman of the Feasibility Evaluation Committee and the members of the Committee are to advise me of their estimations of the various parts of the design. The proposed dates of the meetings and the subjects to be discussed are as follows:

January 16	Primary circuit layout and fuel handling
January 18	Control mechanism and reactivity coefficients
January 20	Size of core and blanket, heat transfer, and shielding of reactor, including biological shield
January 23	Fuel element
January 25	Pumps and emergency and shutdown cooling
January 27	Remote fuel fabrication
3:00 PM February 1	Fuel decontamination
February 3	Fuel cycle, economics, and holdup.

It is not expected that all members of the Committee will be able to attend every session. There does not seem to be any other suitable mechanism for the review of a project which cuts across a number of divisions. Your agreement to serve on this committee will be very much appreciated. Please let my office know if you can accept the assignment.

W. H. Zinn

W. H. Zinn

WHZ:fma

ATTACHMENT 11

MEMORANDUM: L. J. KOCH TO W. H. ZINN, FEBRUARY 10, 1956

CLASSIFICATION CANCELLED
BY AUTHORITY OF THE U.S., A.E.C., C.O.
BY Hoylande D. Young
DIRECTOR OF TECHNICAL INFORMATION
ARGONNE NATIONAL LABORATORY
DATE 2-2-59 D. Adams

CONFIDENTIAL

ARGONNE NATIONAL LABORATORY
Lemont, Illinois

F-200
Koch

CRDC-578

To: W. H. Zinn, Chairman, EBR-II Feasibility Evaluation Committee

February 10, 1956

From: L. J. Koch

102

Subject: EBR-II Feasibility Evaluation

Reactor Engineering

Several questions have been raised by the EBR-II Feasibility Evaluation Committee with regard to specific design considerations and "state of knowledge" pertaining to the EBR-II. In this memorandum I will attempt to cover the more pertinent items and include additional information which may have been developed, or outline proposed programs of development. The questions raised can be grouped into two categories:

1. The basic reactor concept.
2. Specific details of the reference design.

Two major questions were raised with regard to the reactor concept: (1) reactor size and enrichment, and (2) oxide versus metal as the fuel.

Reactor Size and Enrichment

The question was raised as to the maximum enrichment permissible in the reactor to achieve a negative Doppler temperature coefficient in the fuel. The limiting enrichment (if one actually exists) is not known, and therefore it is impossible to establish a design basis. It is planned to investigate this matter as a part of the fast critical experimental program, but there is no assurance that a reliable answer will be obtained.

In the absence of such information, it is proposed to establish a range of reactor sizes for the EBR-II which can be accommodated by the reactor system and the supporting facilities. It is assumed that the total power level of the reactor will remain constant, and therefore the thermal performance of the reactor will be reduced as the core volume is increased. This assumption will eliminate the need for any major changes in the external reactor system.

The largest practical size for the reactor core would appear to be a radius including the first row of the internal blanket subassemblies. The minimum reactor size would be established as the present reference design, including a central blanket. There are an infinite number of possible intermediate size reactor cores between these two extremes. A few have been selected and will be enumerated later.

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The largest core will consist of 91 fuel subassemblies which results in a core with an equivalent diameter of 23.2 in. If an L/D = 1 is established as the maximum core length, then the maximum core volume is equal to 153 liters. This is a very large reactor, and if constructed would fall far short of the original thermal performance goals established for the EBR-II. Since the power density in this reactor would be considerably lower than the reference design (less than one-third), it will be possible to increase the volume per cent of the core devoted to fuel alloy. It is estimated that perhaps 37% of the core volume could be occupied by fuel. The combination of larger core volume, and larger volume percentage of fuel will result in the minimum possible enrichment. It is hoped that it will not be necessary to consider this reactor for construction; however, it will be used as a design basis to determine the requirements for the coolant system and the blanket design for the largest possible core configuration.

The following tabulation lists five possible core geometries, with pertinent data. No. 1 is the large reactor described above, while No. 5 is the reference design with a central blanket. (Only No. 5 contains a central blanket.)

	No. of * Assemblies	Equivalent Dia.-in.	Core Length-in.	Volume ** (liters)	Fuel Vol. %	Critical Mass	Enrichment
1.	91	23.2	23.2	153	37	310 Kg	29%
2.	91	23.2	17.5	113	37	260 Kg	32%
3.	73	20.8	14.2	70	32	210 Kg	49%
4.	61	19.0	14.2	57	32	190 Kg	54%
5.	54	19.0	14.2	50	32	190 Kg	62%

NOTES:

1. (*) The number of assemblies includes the 12 control rods.

2. Core geometries No. 3, 4, and 5 employ reference design fuel assemblies; only the number and enrichment are changed.

3. (**) The core volumes are based on an arbitrary assumption that only one-half of the control rod volume is a part of the active core volume. (The control rods contain only 67% as much fuel as the fuel assemblies. Therefore, the 12 control rods are considered equal to 4 fuel assemblies for this purpose.)

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Many more core volumes can be accommodated within the maximum geometry selected (91 assemblies). The greatest flexibility is produced by varying the core length which is accomplished by changing the length of the fuel pins. The most obvious radial core geometries were selected in establishing these configurations.

The control rod positions, and the length of control rod travel (14 in.) will not be changed in these various geometries. The entire fuel process cycle is being designed to accommodate the 14.2 in. long fuel pins, and it may not be possible to cycle the longer elements in the EBR-II Process Plant. (It may be possible to disassemble and decontaminate the elements, but quite unlikely that the fabrication and assembly can be accomplished.) The construction of the larger cores (core lengths greater than 14.2 in.) should be considered as "experiments" possibly requiring "cold material" and conventional processing.

The reactor and reactor structure will be designed to accommodate any of these core configurations. If reliable information cannot be obtained from the critical experiments, it may prove to be desirable to resolve the question of Doppler coefficient in the reactor. Obviously, this would be a very expensive series of experiments, but future interest might justify the expenditure. In any case, the reactor will be capable of accepting this variety of loadings. This may result in some sacrifice in performance because of considerations of blanket design and blanket cooling to encompass the variety of core configurations. It is probable that over-cooling in some instances will result.

Oxide Versus Metal Fuel

The question of improved economic performance with oxide has been raised. The major improvement is attributed to the high burnup which is believed to be attainable with oxide fuels. Very little direct information is as yet available; however, there are indications that very high burnups may be attainable.

There is reason to doubt the feasibility of employing oxide fuels for small fast reactors. Earlier estimates indicate that there is little difference in critical mass between metal and oxide loadings in small reactors of the EBR-II size. (This is due to the fact that there is little 238 in the metal system.) The EBR-II reference design could not be made critical with oxide (an enrichment over 100% would be required with an oxide density of 10). There are indications that the "inexpensive" oxide element has a density of the order of 6 or 7, which can be obtained without pressing and sintering, etc. A large core volume is indicated even for a fully enriched system.

There are certain disadvantages associated with oxide as fuel which cannot be ignored. If enrichment is a major consideration (Doppler coefficient), then the problem is more serious with oxide because of the lower density. In addition to the considerations of Doppler coefficient, the thermal expansion coefficient of oxide is small, and probably "unreliable" under the conditions of operation. It is quite likely that thermal expansion of the oxide itself cannot be relied upon as a fast shutdown mechanism. In conjunction with a possible

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positive Doppler coefficient this may result in a very undesirable combination. The "volume temperature coefficient" of the reactor would be far too slow during fast transients.

High burnup in oxide fuels introduces the problems associated with containment of the fission gases at high pressures. To contain the fission gases may require a completely new concept of our fuel element design. Earlier estimates have indicated that a void volume greater than the fuel volume is required to contain the gases at acceptable pressures at operating temperature. It does not appear possible to provide this volume adjacent to the core because of the large gap which would be created between the core and the blanket. Combining the core and blanket in a single tube has been considered, (with the expansion volume provided outside the blanket), but this will undoubtedly complicate the assembly of the core section since very long fuel tubes would be required. Assembly of the long, small diameter pins would be difficult, and coolant pressure drop would be increased tremendously. The manifolding of the gas to an external reservoir appears to be a very difficult problem in the pin type element. It does not appear to be feasible to vent the gases to the sodium system unless a very high degree of system contamination can be tolerated.

It may be that the "radiator type element," incorporating the wafer configuration" may be an application for oxide fuels. The powdered metallurgy method of compacting wafers may be applicable to oxides which are also handled as powders. The radiator geometry lends itself to venting of fission gases, since the entire volume of a subassembly can be vented by a single tube, and there is no manifolding problem. (These statements should be recognized as pure conjecture at this stage of development, but I would like to suggest that some of the longer range work we are doing may have application to oxide fueled reactors should they really prove to be superior.)

Because of the very low thermal conductivity of oxide, very high temperatures and large temperature gradients must be accepted. The test results obtained thus far have not indicated any serious problems associated with these high temperatures; however, considerably more experience would be necessary to develop a feeling of confidence in this situation.

As a longer range consideration, the mixed plutonium-uranium oxide may represent a solution for the plutonium fueled reactor. Only one experiment has been reported to date on the mixed oxide, which was the successful irradiation of a 5:1 mixture to a burnup of 5% of the plutonium (.83% total atoms). This sample was of 65% density, and on a volume basis would represent an equivalent burnup of .31% total atoms in metal. KAPL has postulated that a burnup of 50% of the plutonium atoms may be possible. We are in no position at the moment to duplicate equivalent burnups in plutonium-uranium metal alloys, and if they are successful (and we are not) it would appear that the mixed oxide may have real potential.

I would recommend that the design basis for the EBR-II plant continue to be a metal fueled reactor. I believe that a modest effort should be undertaken at Argonne to investigate the feasibility of oxide fuels. This should

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include fabrication, performance, and processing. The mixed uranium-plutonium oxide system is of particular interest. The EBR-II reactor should be included as one of the tools employed to evaluate oxide fuels, including nuclear properties, as well as engineering properties.

Details of EBR-II Design and Performance

Many questions were raised concerning specific design features of the reactor and reactor system. Most of these indicated the need for more intensive analysis or experimentation. I will discuss those items which appeared to "stimulate the most vigorous discussions, or were not reasonably well settled."

Reactor Cooling After Shutdown

Reactor cooling after shutdown can be divided into two categories: (1) "emergency cooling" after shutdown, which includes the short term cooling immediately after shutdown under various conditions of operation, and (2) "shutdown cooling" which involves the long term cooling to remove the fission product decay heat.

The shutdown cooling is reasonably well in hand, in that the heat is removed from the reactor by natural convection of the sodium and the only question involves the reliability of the various means of removing the heat from the primary system sodium. In addition to the intermediate heat exchanger which is the normal "heat sink" during operation, shutdown coolers are provided which operate entirely by natural convection. In addition, there are other methods of removing heat which are available but are not normally on immediate stand-by status. Included are the heat exchange system associated with the sodium cold trap circuit, and ducting which is available for air cooling the primary tank. Since several days are available to initiate the necessary cooling mechanism, it does not appear to be necessary to have a variety of cooling methods available on an immediate stand-by basis. This matter will be resolved on the basis of operational judgment to determine the availability of the various auxiliary cooling procedures.

Emergency cooling of the reactor is not as well established. It is somewhat more difficult to define because of the possible variations in operating conditions which influence the emergency cooling requirements. These include the operating history of the reactor, the position of the control rods, the cause of the emergency shutdown, and the hydraulic characteristics of the sodium cooling system.

The cooling requirements are being established for various types of emergency shutdown, which will be employed to determine the coolant flow requirements. The heat generation decay curves have been determined for various operating conditions, including different amounts of negative reactivity insertions based on various control rod velocities. These were determined by calculating the control rod displacement versus time curves and applying them

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to scram conditions occurring from various control rod positions. The power decay curves were obtained on the REAC and the AVIDAC for various instantaneous negative reactivity additions. The analysis of this problem is still in progress, but it appears that the reactor will cool under the condition when the reactor and pump are "scrammed" simultaneously at full power. There are other conditions which are more severe, and these are yet to be investigated. The most serious condition probably occurs when the pump fails which, in turn, initiates the reactor scram. In this case, the coolant flow decay leads the power generation decay of the reactor, and the cooling problem may be more severe.

In the event that the present coolant flow decay characteristics are inadequate to cool the reactor under all anticipated conditions, various methods of augmenting coolant flow with "inertia systems" will be investigated. If these are also unsatisfactory, then a multiple pump system with separate power supplies will be explored. A layout has already been completed which incorporates two pumps in parallel, replacing the single pump now employed. A design has been prepared which permits the use of either D.C., electromagnetic pumps or mechanical pumps interchangeably in the system by the use of a suitable adapter. The two-pump arrangement simplifies some of the structural problems, because the units are smaller and the space available is more than adequate. In fact, it is possible that the size of the primary system tank can be reduced if two pumps are employed.

The only unresolved problem with regard to parallel pumps is related to possible by-pass flow through an inoperative pump loop in the event of failure. Various systems are being investigated to attain a reliable "check valve system", including the use of shaped nozzles which was suggested during the meeting.

Fuel Element Feasibility

The fuel element feasibility was questioned because of the lack of information on the particular alloy which the fuel process cycle imposes on the reactor system. It was indicated that the objective of 1% burnup is inadequate, and that our sights should be directed to a minimum of 2% burnup. We have some evidence that 2% burnup is attainable in other alloys (notably, the 2% zirconium alloy), but even here the experiments have been conducted on rather small samples.

To improve our position with regard to "demonstrated performance" of the EBR-II fuel element, the irradiation and thermal cycling program on "fissium" will be accelerated.

Fissium is defined as a synthetic alloy which approximates the equilibrium composition of the cycled EBR-II fuel. It cannot be established precisely, because the composition is affected by the fission product yields, and the cycle efficiency. (The "drag out" per cycle affects the equilibrium concentration levels.) Two fissium alloys have been adopted for investigation and are referred to as "3% fissium" and "5% fissium." Their compositions are as follows:

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	3% Fissium	5% Fissium
Zr.	.1%	.2%
Mo.	1.6%	2.5%
Ru.	1.2%	1.5%
Rh.	.2%	.3%
Pd.	.1%	.5%
Total:	3.2%	5.0%

Both of these alloys are under investigation, but it has been decided to concentrate the bulk of our future effort on the 5% fissium alloy. At present, a series of "MTR pins" of both alloys fabricated by various methods and with various heat treatments have been prepared. One fuel element of each alloy is under irradiation in CP-5. Both alloys have been thermal cycled in the alpha range and in the beta range. The 5% alloy is superior under thermal cycling in both temperature ranges. The 5% alloy compares very favorably with the molybdenum alloys under beta cycling which is quite encouraging.

The future program will be confined almost entirely to the evaluation of the as-cast 5% fissium alloy. The only fabrication variable to be explored is the effect of heat treatment on this material. This is a "mixed phase" alloy, and it is desirable to determine whether the fabrication process should be designed to provide the alpha phase or the gamma phase preferentially. Heat treatment can accomplish this, but it is necessary to determine which is the most desirable composition, and whether or not the composition is retained after irradiation.

The major emphasis in the irradiation program will be to obtain high burnups in as short a period as possible, and to irradiate material in the alpha temperature range and in the beta temperature range. The irradiation program will include the standard MTR pin irradiations, the CP-5 fuel element irradiations, and the addition of fuel element irradiations in MTR. The next two fuel elements for irradiation in CP-5 will be fabricated by casting into Vycor tubes. These will be of 30% enrichment and will be irradiated in CP-5. It is hoped to get significant burnup on these samples before the reactor power level is increased. These samples should be under irradiation before the end of this month.

One alloy variation will be investigated because of the possibility that a higher alloy concentration may be beneficial. This may particularly be true if retention of the gamma phase should prove to be important. Two alloys consisting of 5% fissium, plus 2-1/2% molybdenum and 7-1/2% molybdenum will be evaluated in MTR-type pin irradiations. These alloys will contain a total of 5% and 10% molybdenum and will, therefore, extend into the range of high molybdenum alloys.

The following table outlines the present planned irradiation program. It will also be augmented by a flexible thermal cycling program which will be geared to differentiate between thermal effects and irradiation effects on the alloy.

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IRRADIATION PROGRAM

I. MTR Pin Irradiations (Metallurgy Division)

A. Alloys

1. Fissium

~~a. 3% Fissium~~
b. 5% Fissium

2. 5% Fissium plus molybdenum

OK a. Fissium ≠ 2.5% Mo. (5% Mo. total)
b. Fissium ≠ 7.5% Mo. (10% Mo. total)

3. 5% Fissium with impurities from fuel cycle (future program)

a. From cladding (Fe, Cr, Ni)
b. From process or casting (Si)

B. Fabrication

1. As cast - without heat treatment
OK 2. As cast - with "gamma quench" heat treatment
3. As cast - with "gamma slow cool" heat treatment

C. Irradiation Conditions

1. 1%, 2%, and higher burnup
2. Central metal temperatures of 1100F to 1200F
3. Central metal temperatures of 1250F to 1350F

II. Fuel Element Irradiations in CP-5 (Reactor Engineering Division)

A. Alloys

1. 3% Fissium (one sample now under irradiation)
2. 5% Fissium (one sample now under irradiation)
3. 5% Fissium (additional samples)
4. Possibly 5% fissium and Molybdenum (results from I-A-2 above)

B. Fabrication

1. As cast - buttwelded - swaged (now in CP-5)
2. As cast - no working - no heat treatment (being prepared)
3. As cast - with heat treatment (results from I-B above)

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C. Irradiation Conditions

1. .5 to 1% burnup (perhaps higher - later)
2. Central Metal temperature approximately 900F (in CP-5)
3. Central Metal temperature approximately 1100F with cycles into beta
4. Central Metal temperature above 1200F (results from I-C-3 above)

III. <u>Fuel Element Irradiations in MTR (Metallurgy Division)</u>

A. Alloys

1. 5% Fissium
2. Possibly 5% Fissium and Molybdenum (results from I-A-2 above)

B. Fabrication

1. As cast - no heat treatment
2. As cast - with heat treatment (results from I-B above)

C. Irradiation Conditions

1. 1%, 2%, and higher burnup
2. Central Metal temperature to be established (as high as practicable)

The question was raised with regard to the possibility of a progressive type accident occurring if an individual fuel element melted. It was postulated that a fuel element without a sodium bond might be accidentally loaded into the reactor. The fuel element could also melt if a large enough defect was present in the bond. Since the unbonded element represents the most severe condition, a test program is planned to investigate this phenomenon. Three different tests are being considered to evaluate this problem.

1. Molten uranium will be cast into a fuel element tube which is immersed in sodium. In this experiment the sodium may be stagnant or may be flowing. It is planned to examine the static sodium case first, since it is the simpler experiment. Molten uranium will first be poured into a single tube to determine if the uranium will melt through the tube wall. If it does, a cluster of dummy elements will be placed around the tube in the proper locations to determine the final configuration of the molten metal. Depending upon the results, it may be necessary to repeat this experiment with flowing sodium because of the possible effect of the sodium velocity on the distribution of the molten metal. This experiment has the obvious shortcoming in that there is no addition of heat to the metal after it enters the fuel tube. Superheating the metal will partially compensate for this defect, but it is not clear that it does duplicate what may occur in the reactor.

2. The possibility of electrically heating a fuel element in sodium is also being investigated. This will require a very high temperature heat

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source in the center of the uranium pin which will melt the uranium and continue to add heat after the uranium has melted and solidified in its new configuration. It is possible that a tungsten resistance wire insulated from the uranium can be used, and it is being investigated. If successful, this would have the advantage of permitting the approximate duplication of reactor conditions. If the test is conducted in a flowing sodium system, the conditions may be almost identical.

3. An in-pile test is being considered in which the fuel element is permitted to melt within a sealed capsule. This test can only employ a static sodium system, but obviously can duplicate all of the other reactor conditions. This type of accident has already occurred in our irradiation tests, but we would not propose to perform this experiment until some of the above-described work has been completed, and we are better able to predict the results. This would be a very short term test and would not require the elaborate capsule design now employed in our long term irradiations. The activity levels would be much lower, and the disassembly and examination of the samples after irradiation would be much easier.

The possibility of the fuel elements bowing due to a temperature difference across the subassembly was discussed at considerable length. The major question here is the actual temperature difference which will exist across the subassembly in operation. The variation in heat generation across the subassembly can be calculated quite easily, and the bowing of the tube can be calculated if the temperature difference is known. The difficulty lies in our inability to calculate the temperature difference due to the fact that mixing occurs, and that radial heat transfer occurs within the subassembly. These two factors tend to reduce the temperature difference, but it is impossible to calculate these effects. It is proposed, therefore, to set up a full scale experiment to determine the actual temperature difference in the subassembly tube when the heat generation in the fuel elements varies across the tube. Unfortunately, the temperature differences are expected to be quite small (approximately 25F, if there is no mixing) and a reduction in the total power generated in the experiment will result in a corresponding reduction in this temperature difference. It is therefore necessary to build an experiment which delivers a large amount of power in order that the temperatures being measured are not too small. A preliminary design of this experiment has been prepared, and an attempt will be made to deliver at least 500 kw of power to the sodium. This experimental unit will be tested in the sodium pressure drop loop which is now under construction and which has the capacity to pump sodium through a subassembly at rated flow conditions. By varying the power generation in the pins across the test assembly, and measuring the temperature distribution in the hexagonal containing can, the necessary information will be obtained. Once the actual temperature distribution is determined, the actual deflection can be determined very easily by calculation and experiment.

This possible effect is important for two reasons. Under equilibrium conditions it may have an effect on the control rod which may bow and give

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difficulty since it is a moving unit. Under transient conditions, all of the assemblies in the reactor may tend to increase their deflection which could have a marked effect on the reactivity of the reactor.

In the event that the temperature differences are significant, and the deflections are larger than can be tolerated, methods of support must be developed. The most promising arrangement appears to be a system of buttons or projections on the subassembly walls at the centerline of the reactor providing essentially metal-to-metal contact in the system. This arrangement would minimize the cumulative effect which is the most critical during a transient. Since essentially metal-to-metal contact is maintained, the problems of self-welding and seizing must be resolved, and also the effect on the fuel handling procedure. If the projection height is approximately half the clearance between subassemblies, then the clearance for fuel handling is effectively reduced by one-half. A mock-up will be constructed to determine if these reduced clearances can be tolerated.

Various systems of creating "anti-bowing" designs have been discussed (both during and after the meeting). To date, no practicable system has appeared, primarily because of the space limitations and the geometrical problems which arise because the subassemblies do not orient identically with respect to the center of the reactor. (In some cases, the flat of the hex is directed toward the center, and in other cases the corner of the hex is directed toward the center of the reactor.) Also, considerable complexity would be added to the fuel handling system because of the necessity to avoid reversing an assembly by 180°.

Other questions were raised by the Committee, but I believe that they can be answered by "more work." Included in this category are: the neutron and biological shield analyses, details of design of the primary tank arrangement, the control rod design, and practically all of the processes in the fuel cycle. Some modifications have been made in the primary tank design which I believe to be improvements and which resolve some of the problems identified by the Committee. We will be prepared to review these modifications at an early meeting of the Committee. It is also worth noting that the Working Model of EBR-II has been in operation at 750F since the first of the month, and all of the equipment is functioning very well. The modifications which were made in the various units appear to have resolved all of the difficulties which were experienced earlier.

It should be emphasized that some of the experiments outlined in this memorandum are still in the planning stage. Further study may prove that the program must be changed. I have attempted to summarize the situation at this time, and to interject some opinion as well as fact.

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LJK:lrg

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ATTACHMENT 12

PRESS RELEASE, MARCH 3, 1958

ARGONNE NATIONAL LABORATORY ANL-PIO-73
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Lemont, Illinois

INFORMATION FOR PRESS, RADIO & TV
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February 27, 1958

FOR RELEASE:
Mon., March 3, 1958
P.M. Editions

AEC AUTHORIZES ARGONNE NATIONAL LABORATORY
TO CONSTRUCT EXPERIMENTAL BREEDER REACTOR NO. 2

The Argonne National Laboratory has been authorized by the Atomic Energy Commission to proceed with the construction of a large experimental breeder reactor at the AEC's National Reactor Testing Station near Idaho Falls, Idaho. A sum of $29,100,000 has been authorized for the design and construction of this facility.

The reactor is to be known as Experimental Breeder Reactor No. 2 (EBR-2). It was developed by Argonne personnel as a successor to the Experimental Breeder Reactor No. 1 (EBR-1), which was placed in operation at the Idaho Falls site in 1951.

Power engineers regard the breeder-type reactor as economically promising because it produces more fissionable material than is burned in the reactor. This reactor, therefore, provides the possibility of using economically and efficiently all natural uranium rather than only the uranium-235 as in some other reactor types.

The EBR-2 is an integral nuclear power plant. It includes a complete fuel processing and fabrication facility in addition to the reactor, heat transfer systems and steam-electric plant. The thermal power rating of the reactor is 62,500 kilowatts. Net electrical power output is rated at 17,500 kilowatts.

APPENDIX C

ADDITIONAL DISCUSSION OF SPECIAL FEATURES

Appendix C

Additional Discussion of Special Features

Three features of the EBR-II design are very unique and warrant additional discussion; they include the fabrication and erection of the primary tank, the manipulation of fuel, and the generation of steam.

PRIMARY TANK DESIGN, FABRICATION, AND ERECTION

The concept of a reactor and primary cooling system submerged in sodium and contained in a large tank was quite revolutionary. In addition to the consideration of the feasibility of the concept, there existed questions of design, fabrication, and erection of the tank in which this concept would materialize.

Its accomplishment involved elegant engineering, excellent planning, and superb execution. It is a story that should not be lost.

The following is a summary of that story told with a few photographs and brief narrative.

No penetrations were permitted below the sodium level which resulted in a complex top structure. The outer tank was free of any connection to the inner tank below the sodium level to avoid any common failure mode. The primary tank was hung from a supporting structure above the tank. This arrangement created a unique erection process and provisions for thermal expansion. The tank was normally at 700°F and the support structure was essentially at ambient temperature.

As mentioned earlier, basic EBR-II construction was accomplished by fixed price construction contracts. The primary tank construction and erection was performed by the Chicago Bridge and Iron Company under a separate contract, independent of the general contractor constructing the reactor building. To make this possible, the Chicago Bridge and Iron Company was given 90 days to perform their work, including unrestricted access to the main floor of the building and use of the building polar bridge crane. The general contractor completed the primary tank cavity and the main floor of the building prior to the commencement of this work.

Figure C-1 — Completion of the primary tank cavity and the main floor of the building. The tops of the six columns upon which the entire primary tank and support structure were supported are shown penetrating from the cavity area.

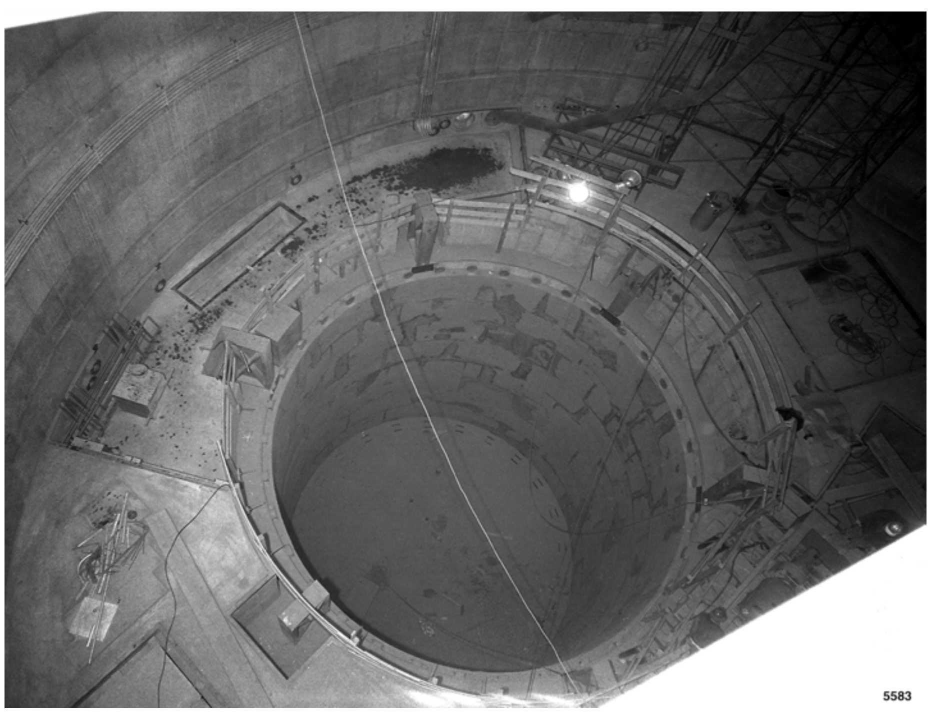

FIGURE C-1. PRIMARY TANK CAVITY.

The total weight of the primary tank exceeded the capacity of the building crane, so a counter weighting system was devised to avoid overloading the crane. Approximately 50 tons of counterweight was needed. Steel balls, 3/8-inch diameter, destined for use as shielding in the primary tank cover, were used as the weight. One of the fabrication problems confronting the project involved the fabrication of the primary tank top cover. The requirement that all access to the interior of the primary tank be through the top of the tank created a complex top cover design containing a large number and arrangements of penetrations. The primary impediments involved fabrication capability, schedule, and coordination.

Figure C-2 — Cover halves were fabricated at two locations in the United States (the Rock Island Arsenal and the Waterton Arsenal). This arrangement involved some risk because the two halves were not connected together until they arrived at the EBR-II Reactor Building.

Figure C-3 — Shipment of the cover halves took extensive planning and coordination. Covers were lifted onto flatbed trailers and securely fastened.

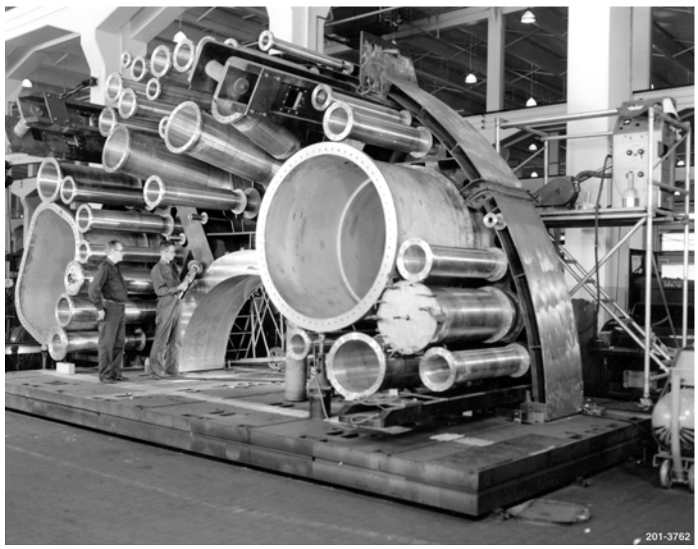

FIGURE C-2. ONE HALF OF THE PRIMARY TANK COVER.

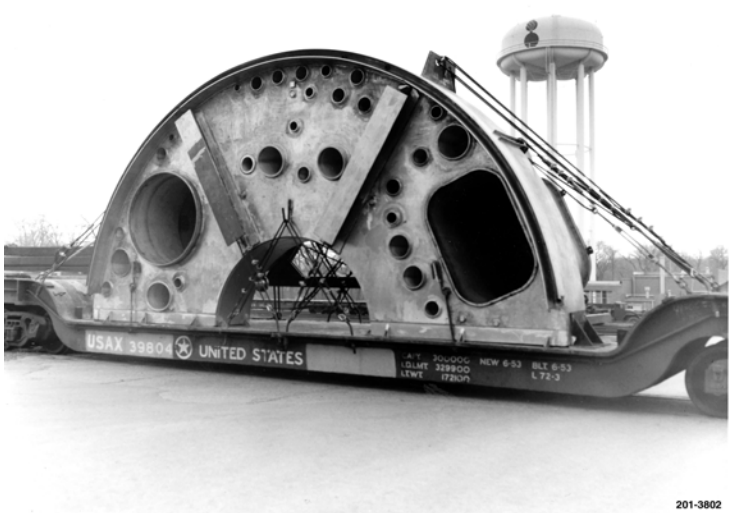

FIGURE C-3. ONE HALF OF THE PRIMARY TANK COVER PREPARED FOR SHIPMENT.

Figure C-4 — Cover halves were transported by flatbed trailer. The weight and size of the load increased travel time. The top cover halves were connected and the inner and outer tanks were fabricated on the Reactor Building operating floor.

Figure C-5 — The work area as the first phase of construction was nearing completion. As depicted in the photo, space was at a premium.

Figure C-6 — The outer (safety) tank positioned in the cavity. The connections between the inner tank and the cover, and the safety tank to the cover were made in the primary tank cavity because of the limited working space available.

Figure C-7 — The inner tank being lowered into the safety tank. These tanks were properly positioned in the cavity in preparation for attachment to the cover.

Figure C-8 — The top cover was supported by the building crane prior to being lowered for attachment to the inner tank.

Figure C-9 — The cover being positioned in preparation for welding to the inner tank. The two tanks are shown positioned in the cavity.

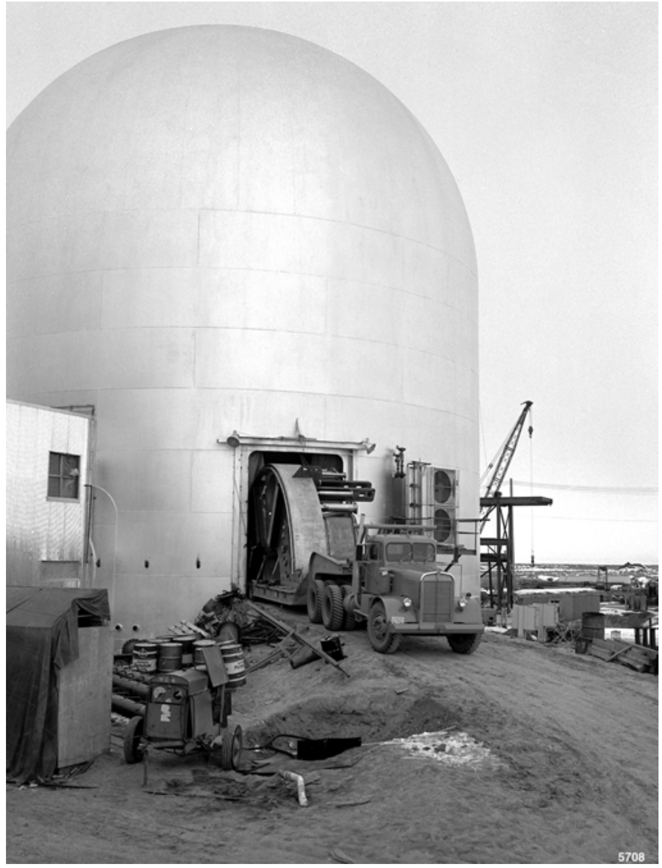

FIGURE C-4. ONE HALF OF THE PRIMARY TANK COVER ARRIVING AT THE EBR-II SITE.

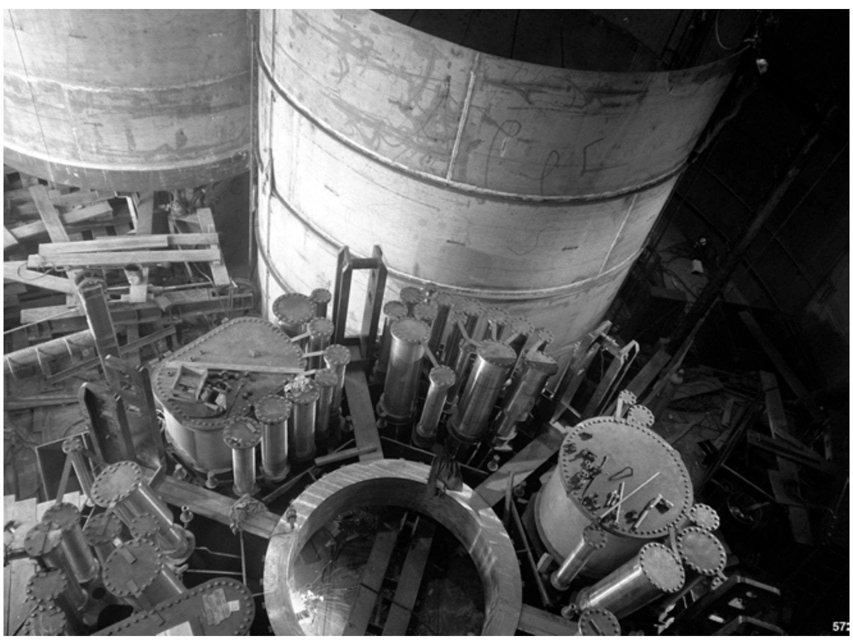

FIGURE C-5. CONNECTION OF THE TOP COVERS HALVES.

FIGURE C-6. OUTER SAFETY TANK POSITIONED IN THE CAVITY.

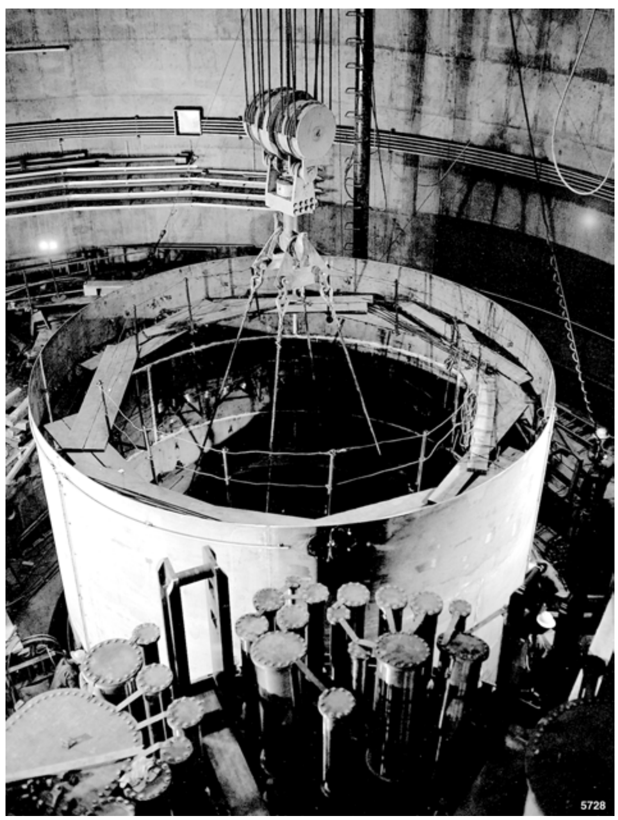

FIGURE C-7. INNER TANK BEING LOWERED INTO THE SAFETY TANK.

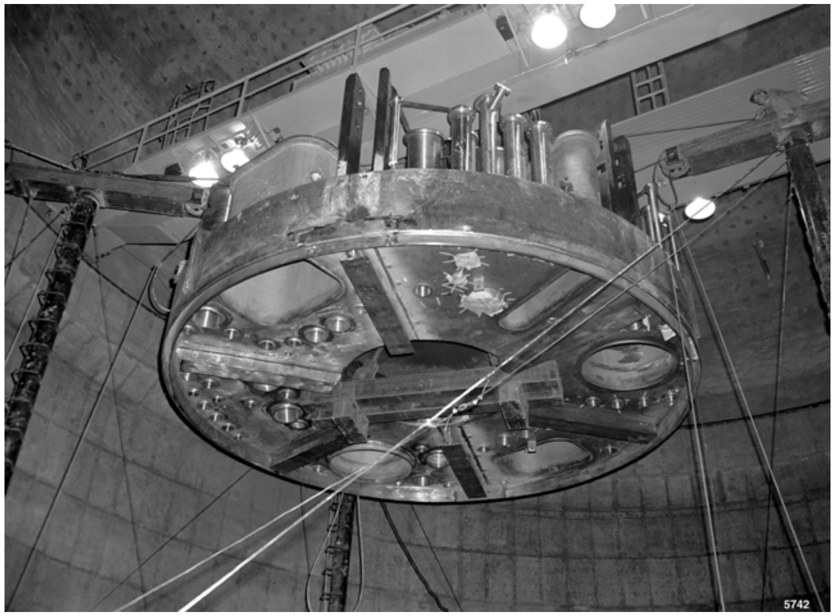

FIGURE C-8. UNDERSIDE OF THE TOP COVER.

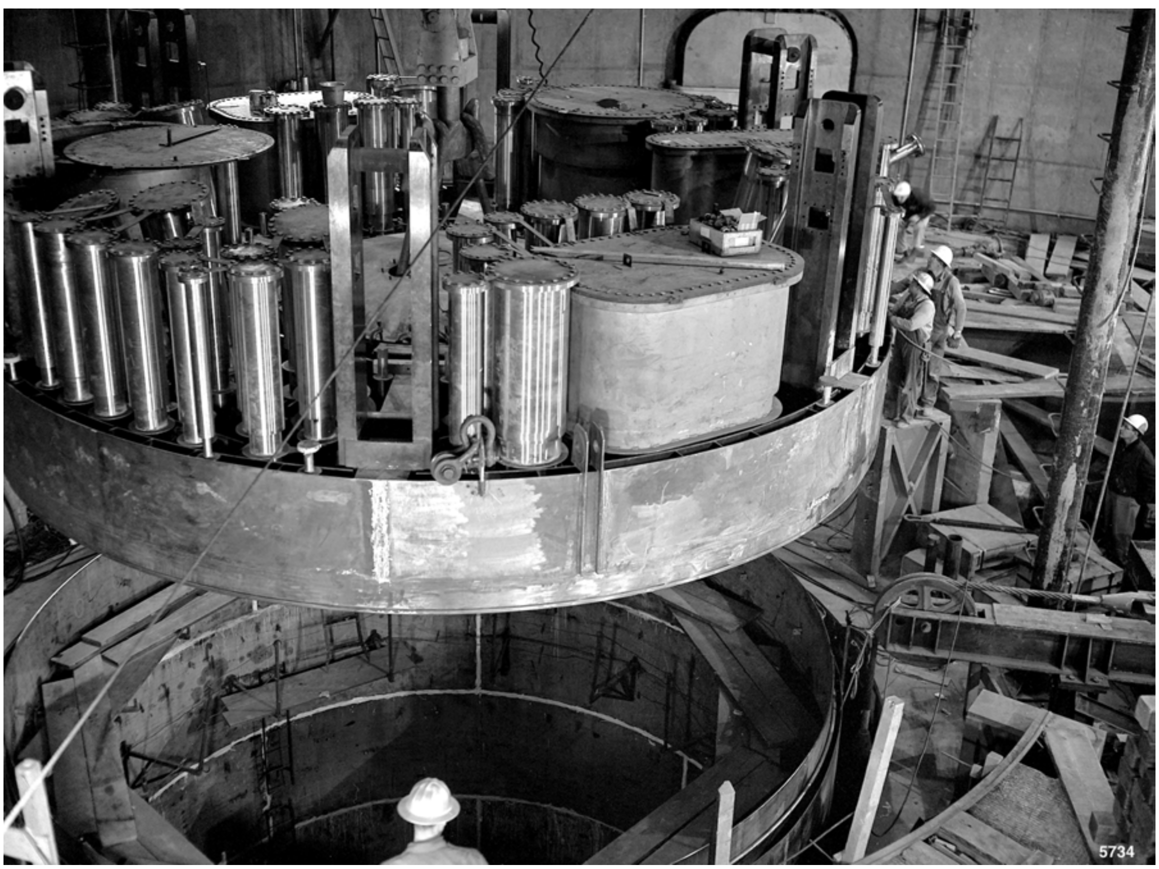

FIGURE C-9. COVER BEING POSITIONED IN PREPARATION FOR WELDING TO THE INNER TANK.

Figure C-10 — The inner tank welded to the cover. The outer tank is shown positioned in the cavity (just below the ladder at the right of the photo). One of the six support columns is shown just to the left of the ladder. One of the counter weights is shown in the foreground filled with the steel balls which were used later for shielding.

Figure C-11 — The inner tank welds were double-butt welds and subjected to 100 percent radiographic inspection. This operation was performed by raising and lowering the primary tank into and out of the cavity to provide appropriate access for performing the work. The inner tank and cover assembly were tested extensively prior to attachment of the outer tank to the cover including helium leak testing of all welds (as shown). The outer tank attachment to the cover was the only single-butt weld and not radiographed.

Figure C-12 — The primary tank with the outer tank attached to the cover is shown supported by the building crane at the center, augmented by the counter weights described earlier. One of the counterweight support columns is shown just to the right of the crane hook and cables, while the counterweight box is positioned directly behind the support column.

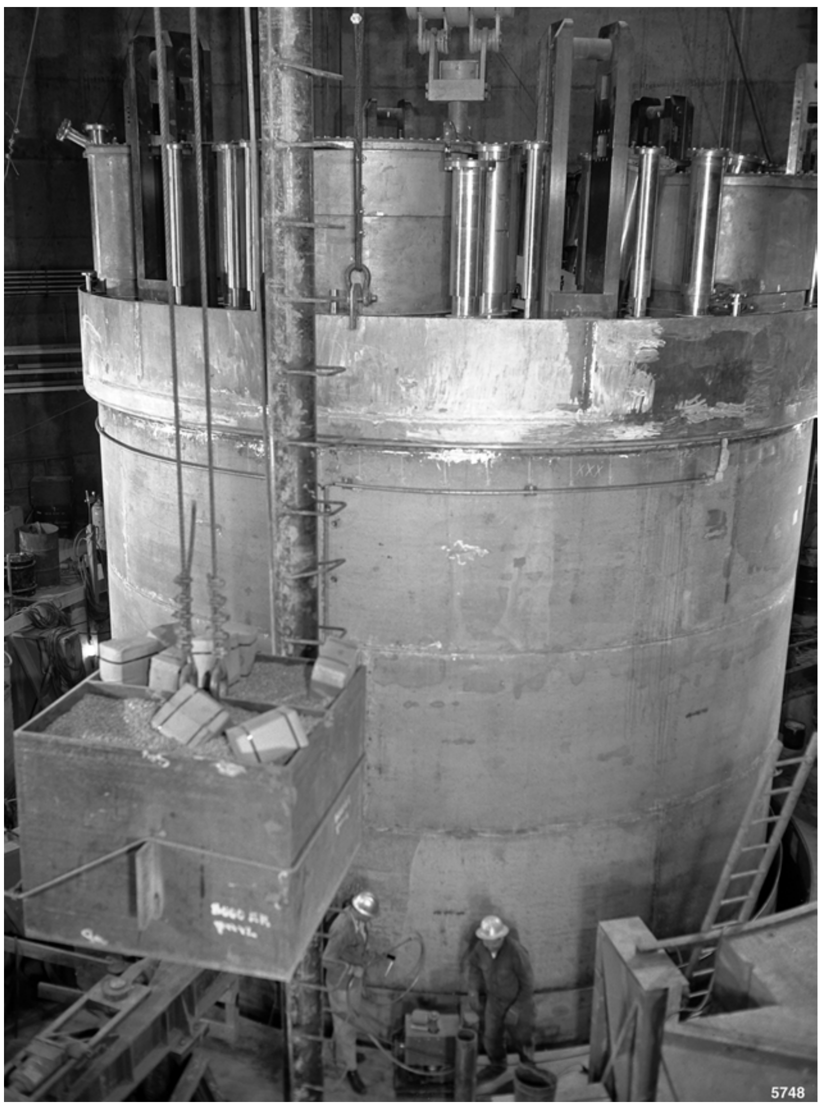

FIGURE C-10. INNER TANK WELDED TO THE COVER.

FIGURE C-11. INNER TANK WELDS BEING LEAK TESTED.

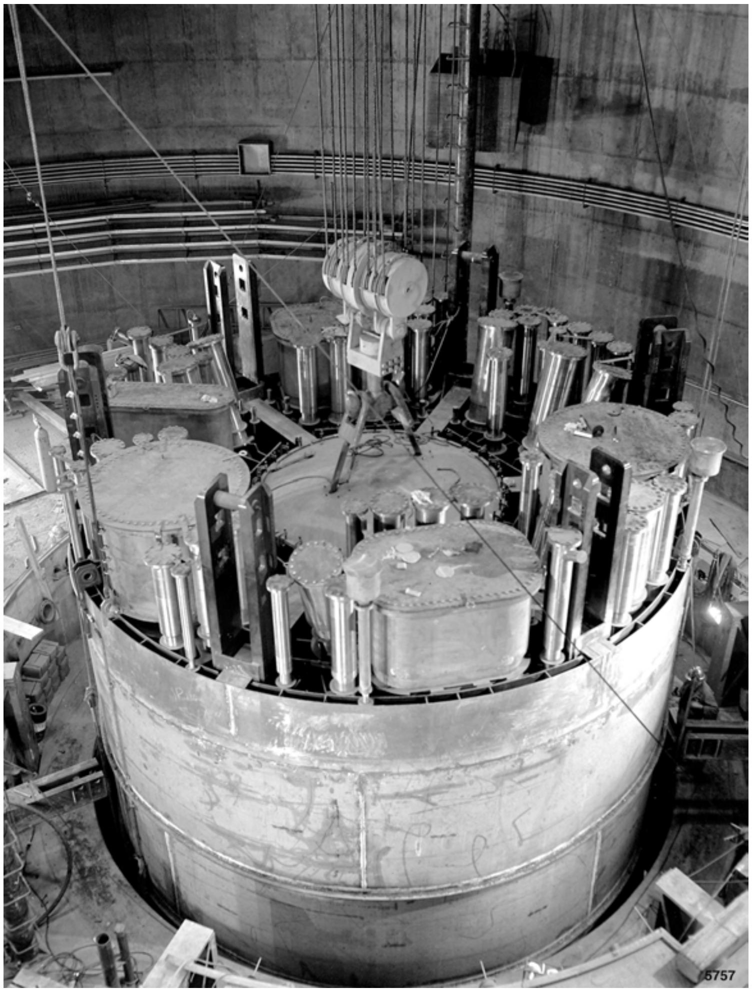

FIGURE C-12. OUTER SAFETY TANK ATTACHED TO COVER.

Figure C-13 — A close-up of the insulation being installed. A technician is installing pre-formed insulation pieces to the outer wall of the primary tank. The beam shown to the right (just below the insulation installer) was one of three used to temporarily support the tank after fabrication was completed and prior to the installation of the upper support structure.

Figure C-14 — The insulation as it nears completion. The counterweight column is shown in the foreground.

Figure C-15 — The primary tank in position in the biological shield cavity. The tank was temporarily supported by the three cable hangers. A gap between the primary tank and the shield cavity was completely sealed to prevent debris from falling into this space.

Figure C-16 — The upper support structure being assembled adjacent to the primary tank. The complete support structure, consisting of the central ring and six beams, were shop fabricated. The beams were then cut off for shipping and then reattached in the field.

Figure C-17 — The support structure was positioned above the primary tank. It was supported by the building crane and "shored" above the support columns.

Figure C-18 — The support structure in place with the six beams attached to the support columns. The primary tank was supported by six temporary hangers at the top of the beams. These six temporary hangers incorporated hydraulic lifts to permit lifting the tank to install the roller assemblies which would be the permanent operational hangers for the primary tank. It should be noted that these hydraulic lift assemblies were designed to be used during the operating lifetime of the plant. The roller assemblies, including the roller and plates, could be removed and replaced by the same procedures if necessary. In this photo, the temporary cable hangers are still in place, but not being used.

Figure C-19 — The top of the primary tank prior to enclosing the upper structure. The primary tank is now hung from its permanent roller hanger assembly. The workmen were standing on the thermal insulation on the top cover of the primary tank. In the next phase of construction, a steel floor for the upper structure was attached to the lower flange of the six upper structure beams.

Figure C-20 — A close-up of one of the six hangers.

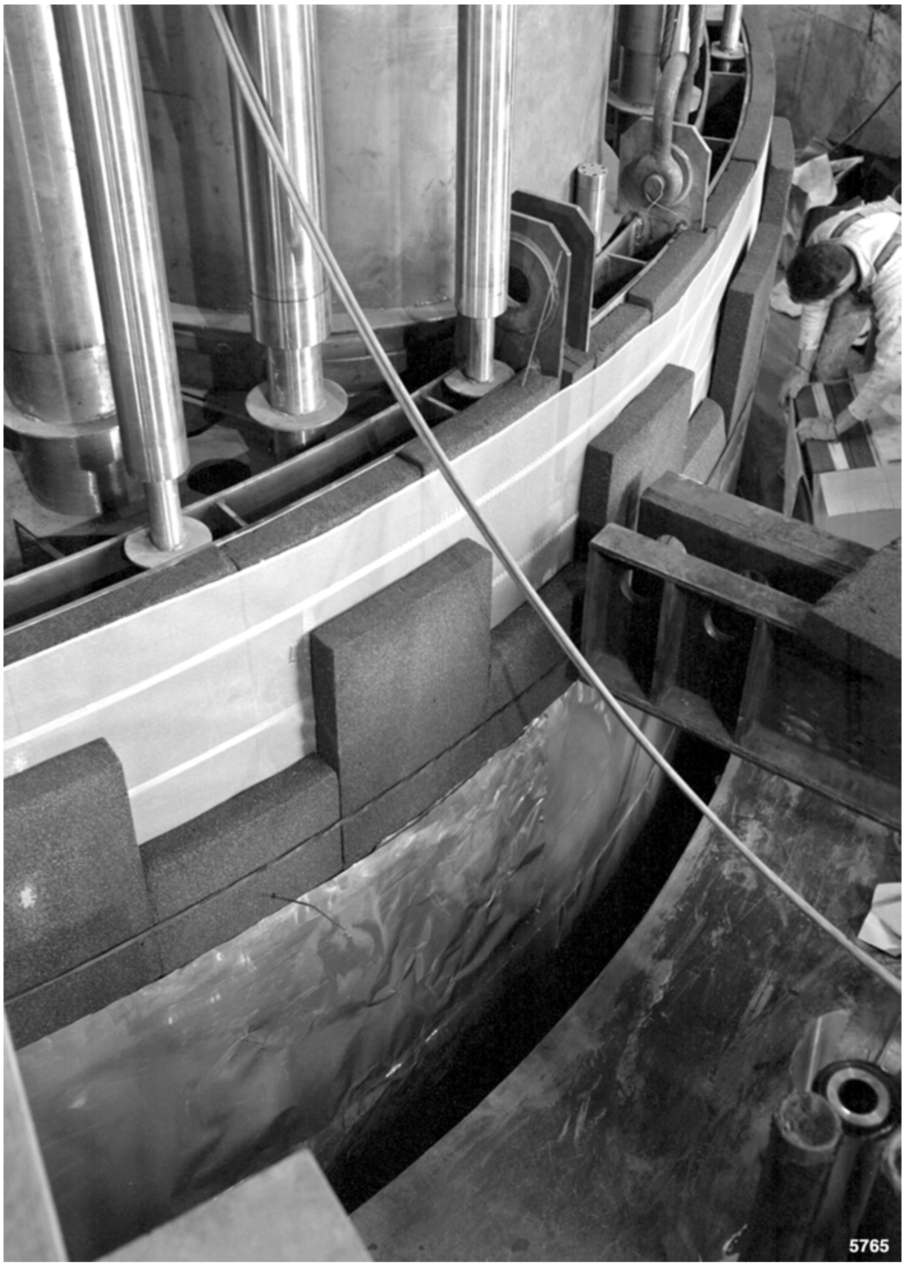

FIGURE C-13. A CLOSE-UP OF INSULATION BEING INSTALLED IN THE PRIMARY TANK.

FIGURE C-14. INSTALLATION OF THE INSULATION NEARLY COMPLETED.

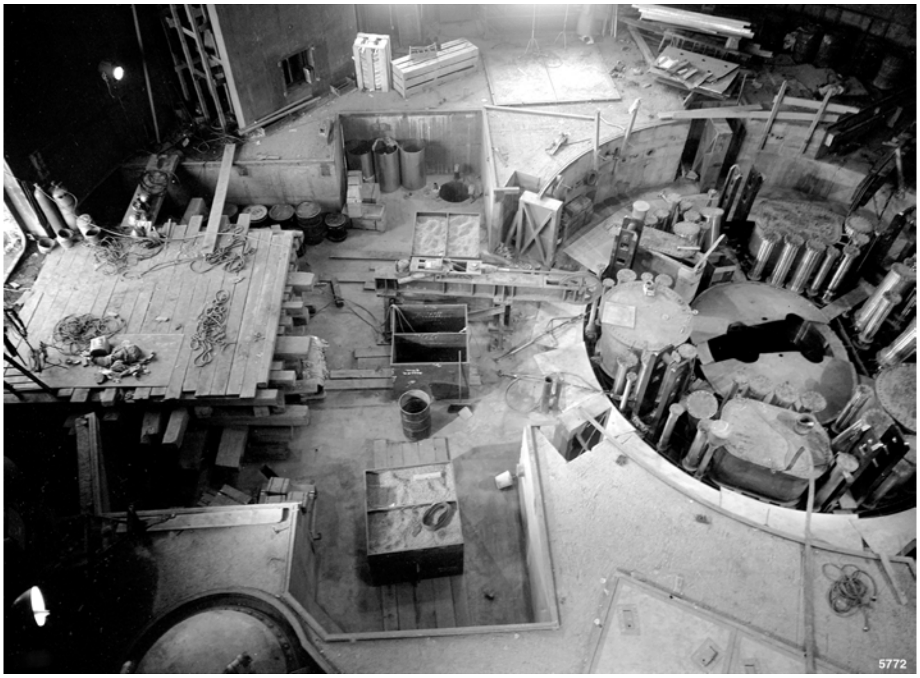

FIGURE C-15. PRIMARY TANK IN POSITION IN THE BIOLOGICAL SHIELD CAVITY.

FIGURE C-16. UPPER SUPPORT ASSEMBLY STRUCTURE.

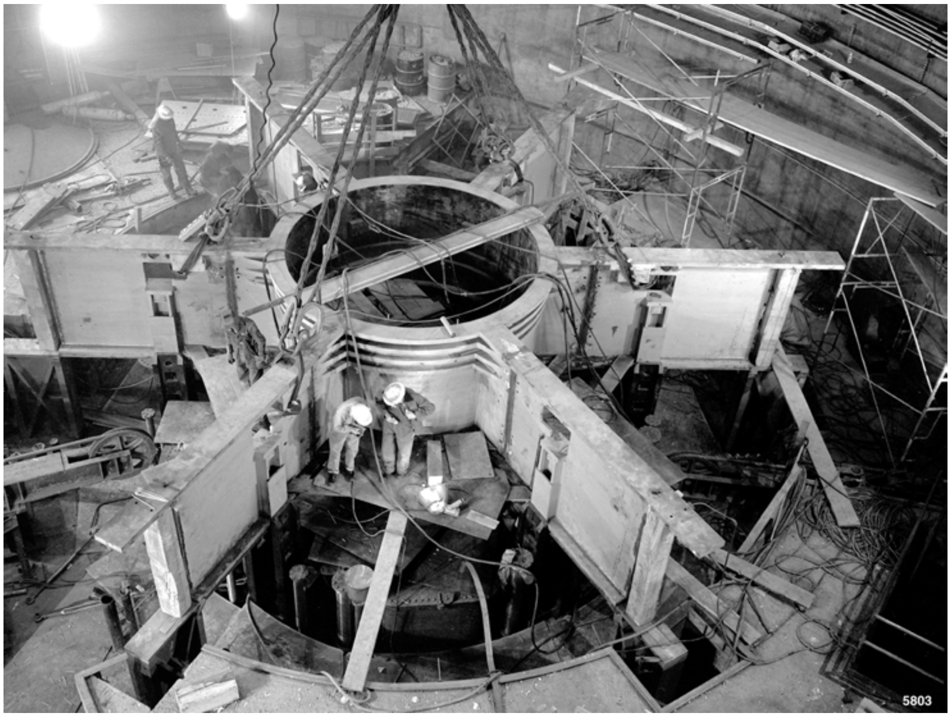

FIGURE C-17. POSITIONING OF SUPPORT STRUCTURE ABOVE THE PRIMARY TANK.

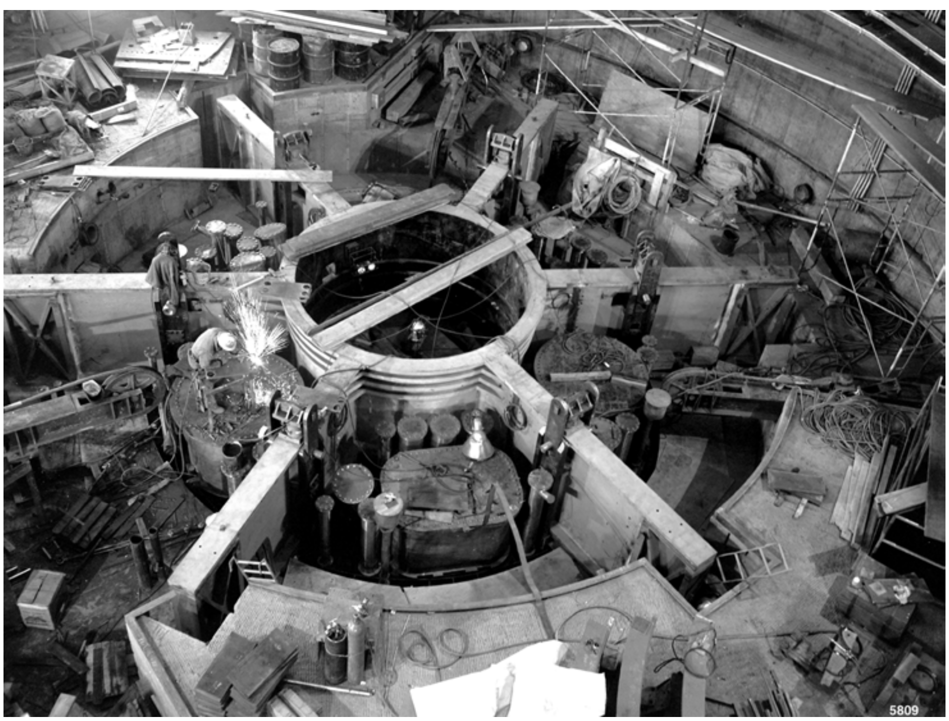

FIGURE C-18. SUPPORT STRUCTURE IN PLACE. SIX BEAMS ATTACHED TO THE SUPPORT COLUMNS.

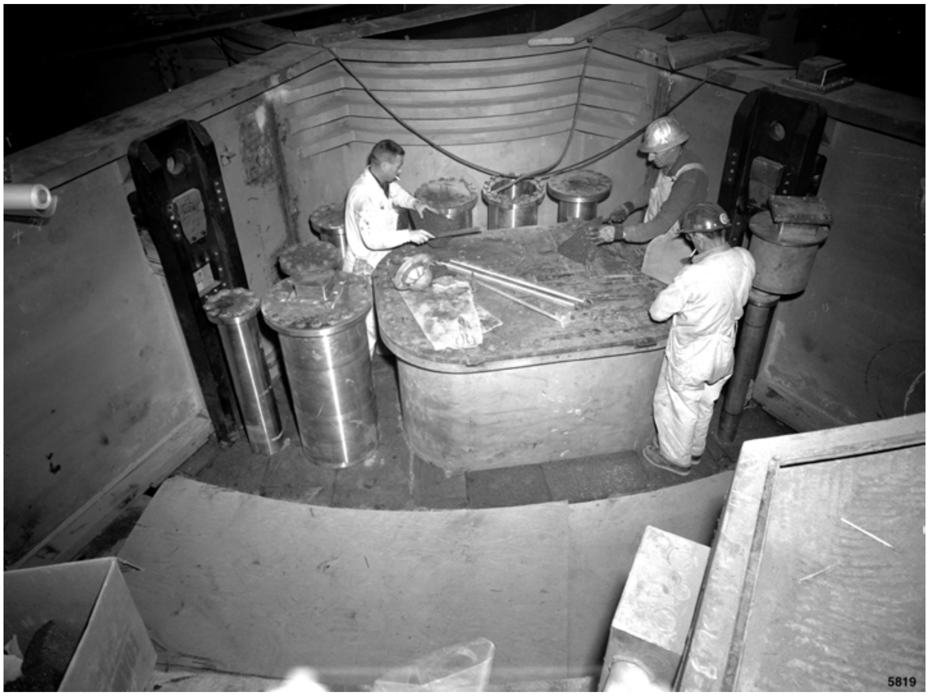

FIGURE C-19. PRIMARY TANK PRIOR TO CLOSING THE UPPER STRUCTURE.

FIGURE C-20. HANGER ASSEMBLY.

Figure C-21 — A sleeve, attached to the floor and surrounding each primary tank nozzle, provided sufficient clearance to permit the nozzle to move freely within the sleeve as the primary tank cover expanded and contracted due to temperature change in the primary tank.

Figure C-22 — These sleeves provided the forms for confining the concrete which filled this region of the upper supporting structure. This arrangement provided access to all of the penetrations into the primary tank and a subfloor work space for equipment.

Figure C-23 — The floor of the Reactor Building. Appropriate removable floor plates were incorporated and positioned at the top level of the supporting structure.

One of the concerns about the EBR-II primary system concept involved the compactness of the system and the clutter that would result from the need to place so many components and related equipment in such a small space. In a larger system, this situation could be improved because of the increased space available. The number of control rod drives, instrumentation, pumps, and heat exchangers would not increase proportionately to an increase in power level. The designers of larger plants should be able to improve on this aspect of the EBR-II design.

FIGURE C-21. SLEEVES IN THE PRIMARY TANK.

FIGURE C-22. CONCRETE AND THE UPPER SUPPORTING STRUCTURE.

FUEL HANDLING, TRANSFER, AND TRANSPORT

The installation and removal of subassemblies into and from the reactor involved a number of complex operations conducted under opaque conditions, 700°F sodium, and in very restricted conditions. Similar procedures performed in water-cooled or gas-cooled reactors permit visibility of the operations and are performed under more favorable conditions.

In recognition of these more difficult requirements, EBR-II operations were based on employing a series of relatively simple operations supported by positive feedback of information to indicate that each was performed properly. In addition, "artificial intelligence" was provided to human operators but "human intelligence" was superimposed on the process. This precautionary approach was taken to partially compensate for the absence of visibility of the operations. Each operation was performed in a positive manner, in accordance with established policies and procedures, and only after prescribed preparatory conditions were met. Upon completion of an operation, feedback was provided to verify that the operation was completed. The same philosophy was applied to operations that were performed mechanically or manually. All operations must be initiated by the human operator; they were not initiated automatically upon completion of the preceding operation even if the feedback requirements were accomplished satisfactorily. The next operation could not be performed if the preceding operations were not complete, even if the operator initiated the action. It is a "permissive" system, but not an automatic system. The system only responded to correct instructions and only if the prerequisites were met.

These operations were controlled directly or supervised by the operator at the fuel handling console located on the operating floor in the Reactor Building.

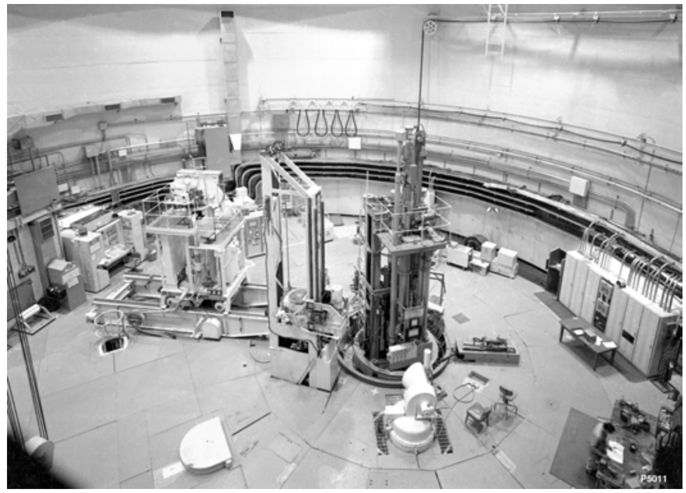

FIGURE C-23. FLOOR OF THE REACTOR BUILDING.

Figure C-24 — The fuel handling console is the primary control point for fuel handling operations. With the exception of the transfer arm, all mechanisms constituting the fuel handling system were controlled and supervised by an operator at the fuel-handing center. The transfer arm was manually operated; however, its operation was limited to the appropriate motions in the fuel-transfer cycle by electro-mechanical interlocks. The position of the transfer arm was transmitted to the fuel handling center.

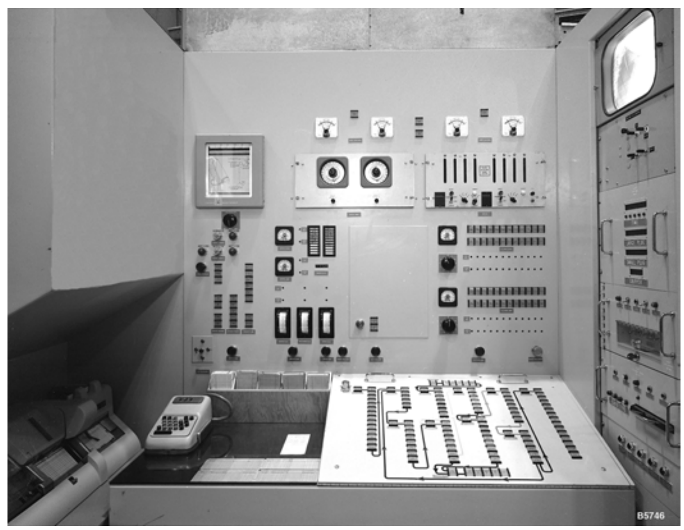

FIGURE C-24. FUEL HANDLING CONSOLE.

Figure C-25 — Functions were conducted sequentially as depicted by the lights and switches comprising the control section of the panel. (These controlled operations applied to both unrestricted operation, with the reactor shutdown and available for fuel handling, and to restricted operation, with the reactor operating.) Key switches in the control room and on the fuel handling console permitted these respective operations.

The sequential pattern of pushbutton-indicator lights depicted the variety of operations involved in preparation for and conduct of the fuel handling and transfer operations. The pushbutton-indicator units were arranged in 12 groups or "operating sequences" as follows:

FUEL HANDLING OPERATING SEQUENCES

Sequence	Function	Sequence	Function
A	Prepare core operation	G	Gripper to core
B	Core to gripper	H	Conclude core operations
C	Gripper to transfer arm	J	Prepare fuel unloading machine
D	Transfer arm to storage rack	K	Fuel unloading machine to transfer arm
E	Storage rack to transfer arm	L	Transfer arm to fuel unloading machine
F	Transfer arm to gripper	M	Conclude fuel unloading machine operation

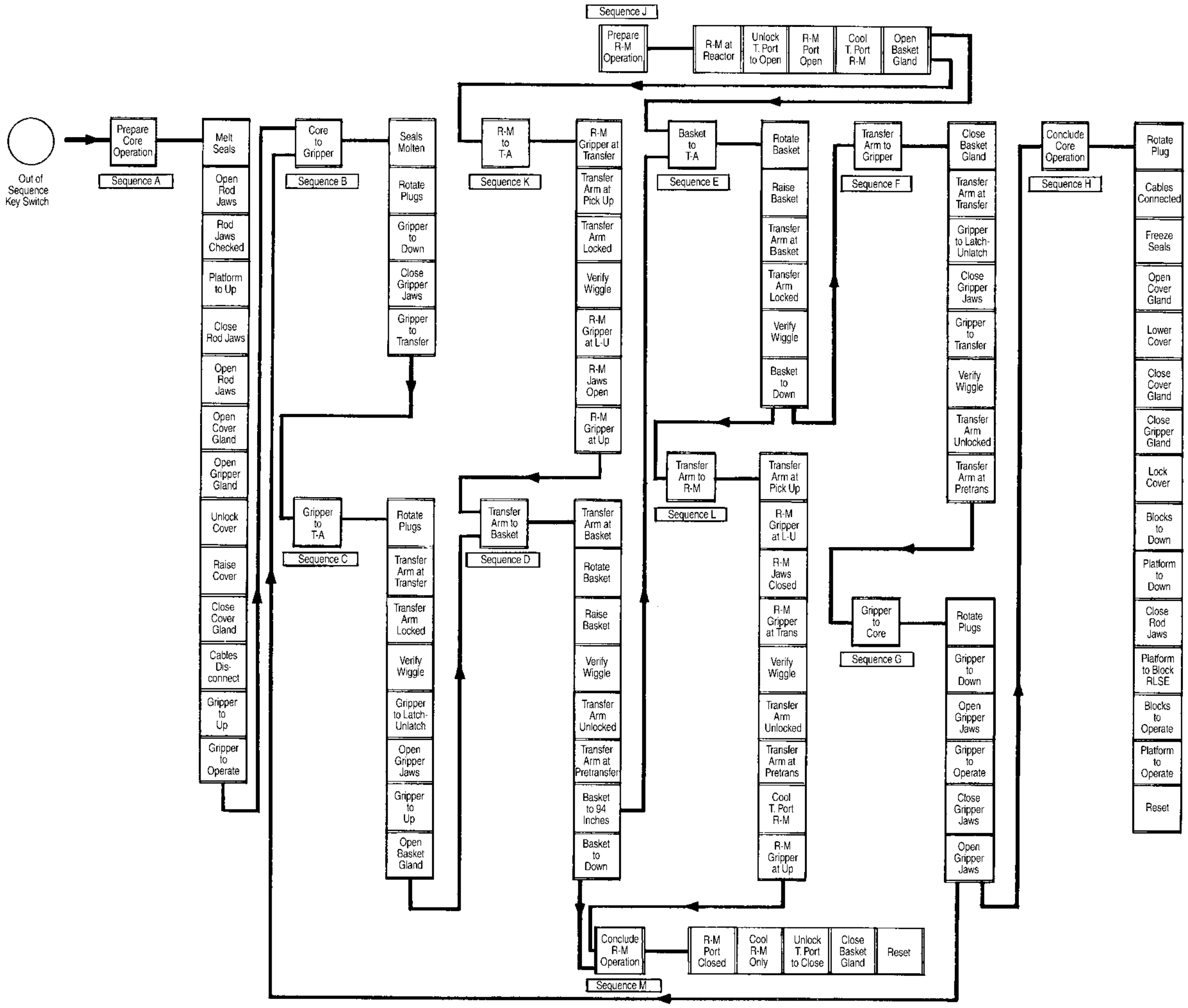

FIGURE C-25. FUEL HANDLING CONSOLE OPERATING PANEL.

Except for the preparatory and terminal sequences (designated Sequence A, H, J, and M), each sequence consisted of a series of steps, arranged in the order of execution essential to the transfer of a subassembly from one location or handling device to another. For example, the steps in the group designated Sequence B extracted a subassembly from the reactor and raised it to the proper level and position for transfer to the transfer arm. Guidelines were provided on the panel indicating the normal progression from sequence to sequence.

To perform the individual operations within a particular sequence, the operator first pressed the corresponding sequence button, which set up the circuits for the operations within that sequence. The operator then successively pressed the operating pushbuttons within the sequence.

Typically, when a pushbutton was pressed, the drive motor associated with the particular motion started and the button illuminated with a red indicating light. Upon completion of the motion, the drive motor stopped and the red light was automatically replaced by a green light. The green light signaled the step was completed, and the following step may be initiated. The indicating light units associated with operations initiating elsewhere (i.e., at the transfer arm or fuel unloading machine), were identical to the pushbutton units, except that the switching function was omitted). The arrangement of controls in operating sequences resulted in some duplication of pushbuttons and indicators, but made it unnecessary for the operator to memorize the sequential order of the many steps. It also provided a more readily comprehended picture of the progress of the transfer operation.

Two key-operated switches were provided on the console for energizing the control power for the fuel handling and fuel-transfer circuits.

1. The “restricted operations” key switch was used for fuel-unloading operations not involving access to the reactor. It energized circuits necessary for operation of the storage rack, transfer arm, and transfer port, provided that the fuel handling control power key switch in the control room was switched on.

2. The “unrestricted operations” key switch energized the control circuits for all fuel handling and fuel-transfer operations. Before this key switch was effective, many interlocks had to be satisfied. In addition, the control room three-position switch, “reactor operate-off-fuel handling” had to be switched to “fuel handling.”

Upon completion of “unrestricted” fuel handling operations, the interlock circuits required the following conditions be met before the control rods were raised to begin reactor operation:

1. The reactor vessel cover must be down and locked to the vessel.

2. The fuel handling operator must have completed the terminating sequence of operations.

3. The “unrestricted operational” key switch on the fuel handling console must be switched to “reactor.”

The fuel handling, transfer, and transport control system involved and required extensive feedback information. The various devices included incorporating sensing capability to provide this information. For example, the gripping concept incorporated into the control rod drive (Figure 3-20) and the fuel handling gripper (Figure C-26) as follows:

1. The gripper jaws were operated positively (i.e., with the jaw activating sleeve up, the jaws are open; with the jaw activating sleeve down, the jaws are closed). Therefore, the position of the sleeve was a positive indication of the jaw position.

2. The sensing device position indicated whether or not a subassembly was in the “gripped position.”

3. If the subassembly was jammed in the gripped position when the sensing device indicated it was not, the jaws would not close.

4. Therefore, the “empty” position of the sensing device and the “closed” position of the gripper jaws were positive verification that the subassembly was not engaged by the gripper.

Figure C-26 — The amount of motion involved in each of these operations is easily detected in the gripper jaw mechanism.

Great reliance was placed on positive position indication of devices and supporting informative components. This information was made available to the operator and to the fuel handling control center. The operations were controlled by the fuel handling control system which incorporated a numerically controlled positioning system. This system was developed in the late 1950s, preceding the development and application of computer chips in control systems.

Similar position and function feedback information was incorporated into the various devices involved. In addition, many of the operations were totally dependent upon the accurate angular positioning of devices located in the rotating plugs, such as the subassembly gripper drive.

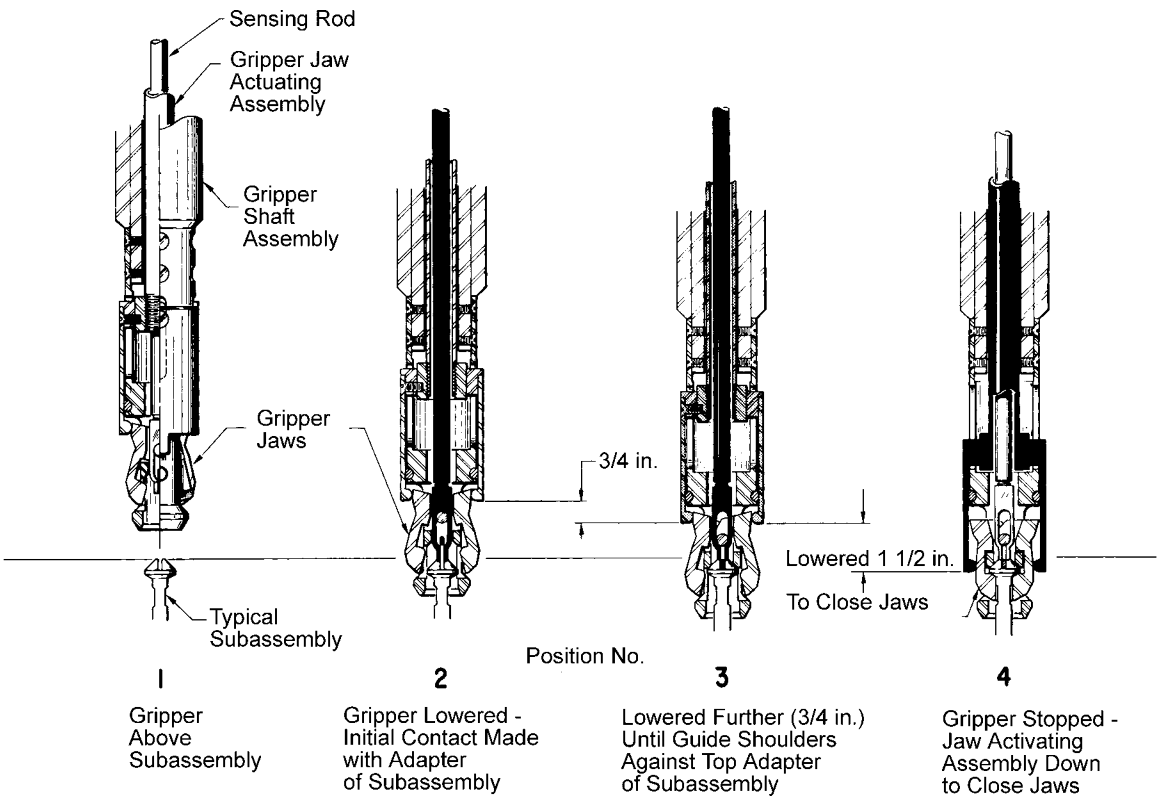

FIGURE C-26. GRIPPER JAW ACTUATING MECHANISM.

The fuel handling control system performed two distinct functions: (1) control operation of the various components in the system, and (2) creation of a record (gather data). Repetitive, identical motions for a given operation or operating sequence were controlled by conventional circuits utilizing limit switches or push buttons. Motions that were a function of the point of origin or destination of a subassembly were performed in response to the numerically controlled positioning system. For example, plug rotation, gripper rotation, storage rack rotation, and vertical positioning were controlled by this system.

Prior to the fuel handling or transfer operations, the following information was supplied to the system on “input” punched cards: (1) type of operation (i.e., reactor to storage rack); (2) card serial number; (3) subassembly identification; (4) core location: row, sector, and number; and (5) storage rack, rotating plug and gripper angular coordinates. The approximate accuracy and resolution of the numerical positioning system are summarized below.

Mechanism	Positioning Accuracy	System Resolution
Large Rotating Plug	4×10^{-5} rev	1.0×10^{-5} rev
Small Rotating Plug	5×10^{-5} rev	1.0×10^{-5} rev
Gripper (angular position)	1×10^{-3} rev	5×10^{-4} rev

The storage rack elevation and rotation drives were controlled in an “on-off” fashion to effect indexing at certain discrete points, the accuracy of which was determined by limit switch settings and mechanical devices. The transfer arm positions over the storage rack were supervised, but not controlled by the system.

The input information for the numerically-controlled positioning system was contained on four types of punched input cards, each of which was identified by color, and a number punched and typed on the card:

Type	Color	Operating Functions
1	Blue	Fuel unloading machine to storage rack
2	Green	Storage rack to reactor
3	Red	Reactor to storage rack
4	Yellow	Storage rack to fuel unloading machine

At specified points in the fuel handling cycle, the appropriate input card was manually inserted into the card reader. The card was immediately read and the information stored. After automatic checks indicated that a valid input card was being used, and in the proper sequence, motion of the rotating plugs, or storage rack, was initiated when the related pushbutton on the fuel handling console panel was pressed. Upon completion of the positioning operation, the coordinates reached were automatically punched on an output card. After completion of all positioning operations specified on a given input card, the input and output cards were visually compared by the operator for agreement between the command coordinates and the actual coordinates.

The completion of an operation directed by the numerically-controlled positioning system was indicated by a green light in the pushbutton unit as in the case of simpler motions where limit switches were used. However, in the former instance, the green light did not appear unless automatic checks indicated that the position was within tolerance, and that the position encoder data were valid.

Several measures were taken to enhance reliability and eliminate errors. Coordinate information was punched in duplicate on the input cards. The two sets of data were automatically checked for agreement before the system took action. Parity checking and redundancy checking techniques were employed, where appropriate, in the position transducer data handling circuits. Finally, each input card was punched with a serial number which was automatically checked for proper sequence.

A permanent record of operations regulated by the numerically controlled positioning system was made on punched output cards. The output cards were automatically punched as the mechanisms were indexed in response to the input cards. The information punched corresponded to that on the input cards except that the coordinate data were derived from actual mechanism positions. The output cards were also punched to record the date and time. Thus, a permanent record of actual performance were available and were checked against the input cards.

Fuel Unloading Machine

The fuel unloading machine (Figure 3-37) was a heavily-shielded electro-mechanical device that received a subassembly from the transfer arm and delivered it to the inter-building coffin. The subassembly was gripped by the fuel unloading machine and lifted from the transfer arm. A long vertical travel was involved in this process and to minimize the height of the fuel unloading machine a system involving a pair of chain link drives was employed. The chain links were flexible in only one direction; the chains could be coiled in one direction, but were rigid in the other direction. When paired in opposite directions, the pair of chains became a rigid rod. This combination served as the gripper drive assembly, performing a similar function to the gripper in the fuel handling system. It is a considerably simpler function, however, since the subassembly was only raised into the fuel unloading machine and later lowered into the inter-building coffin.

The fuel unloading machine was positioned on a carriage that traveled between the transfer port and the inter-building coffin. All of the fuel unloading machine actions were controlled by an operator positioned on the fuel unloading machine. The fuel unloading machine included an argon gas circulating and cooling system to remove fission product decay heat from the subassembly.

Figure C-27 — The operating positions of the fuel unloading machine. The transfer arm is shown in the position for transfer to the fuel unloading machine.

Figure C-28 — The transfer arm in the position for transfer to the fuel unloading machine during construction.

MANUAL OPERATIONS

The fuel handling and unloading procedures included the direct manual operation of certain devices as described below. Interlock circuits required attainment of the correct position before the next operation could be performed.

TRANSFER ARM

The transfer arm was, by far, the most critical manually operated component in the entire fuel handling, transfer, and transport operation. It functioned in both the unrestricted and restricted operation sequences. The transfer arm was the vehicle for transferring subassemblies between the reactor and the fuel unloading machine with or without an intermediate stop at the fuel storage rack.

During unrestricted fuel handling and transfer, the transfer arm was the vehicle for transferring the subassembly to or from the reactor. Transfer could be directly between the reactor and the transfer port, to the fuel unloading machine, or could involve an intermediate stop at the fuel storage rack in either direction. This capability provided maximum operational flexibility of the reactor and the fuel cycle, and was particularly important in accommodating on-site fuel cycle operations.

Although the transfer arm is operated manually, rotation and locking (or unlocking) were physically prevented at other than the proper positions by electro-mechanical locking devices. During restricted fuel transfer with the reactor in operation, an additional blocking device prevented rotation of the transfer arm to points above the reactor vessel. At specific points in its rotation, the actual position of the transfer arm was transmitted to indicators at the fuel handling console.

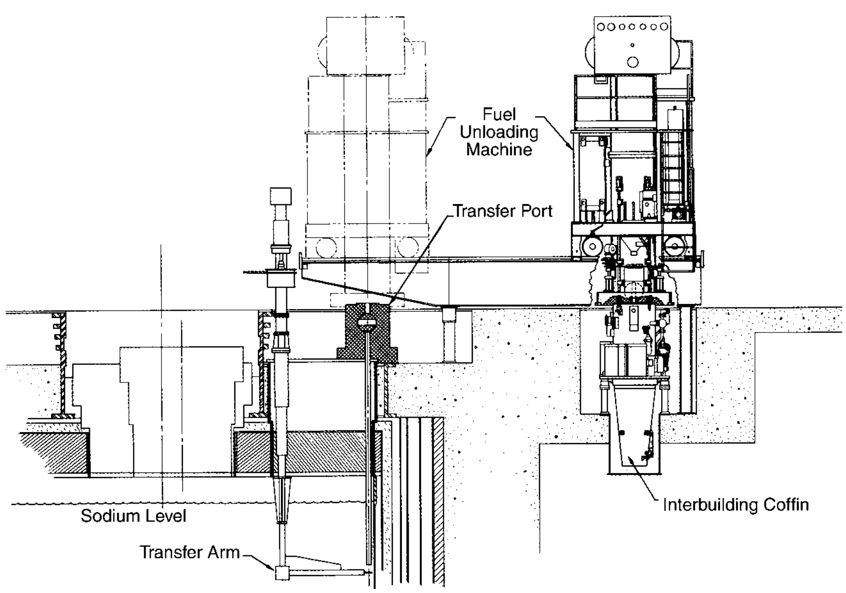

FIGURE C-27. FUEL UNLOADING MACHINE TRANSFER POSITIONS.

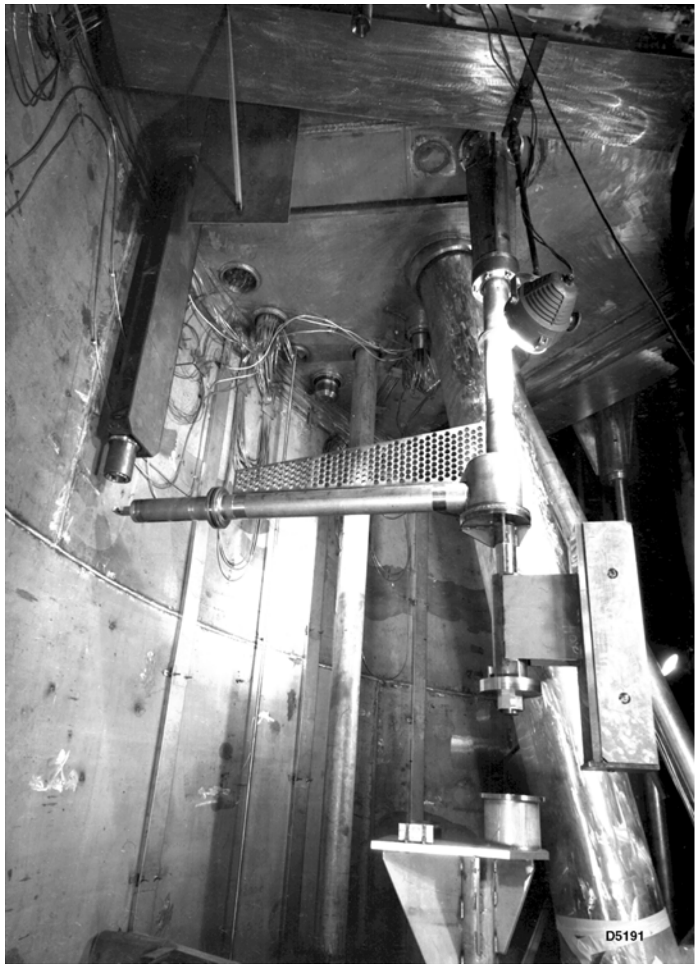

FIGURE C-28. FUEL ASSEMBLY TRANSFER ARM.

The restricted fuel transfer/transport operation is depicted in Figure 3-36 with the reactor shown in the operating configuration. Fuel transfer/transport operations were totally independent of reactor operations. The fuel transport process involved delivering the subassembly to the inter-building coffin and then transporting the inter-building coffin through the equipment airlock to the Fuel Cycle Facility. These operations were typical of those involving the movement of heavy shielded containers involving the use of cranes and transport dollies or carts with one very significant additional requirement. The subassembly required cooling during all stages of this process. The only acceptable coolant to serve this purpose was argon gas which required forced circulation. The continued use of sodium as the coolant after removal of the subassembly from the primary tank would have been totally impractical in the EBR-II concept.

As a result, the fuel unloading machine and the inter-building coffin each contained a rather sophisticated argon gas circulation/cooling system. The inter-building coffin cooling system was further complicated by the need for total energy and operational independence because of the isolation created as the coffin passed through the equipment air lock. This independent system had to be self-sustaining for a considerable period to accommodate the possibility of delay in passage through the airlock. In addition, the inter-building coffin had to be capable of permitting cleaning the subassembly to remove the sodium adhering to the subassembly and had to be capable of accommodating inert gas and/or air as the cooling medium.

These special requirements added complexity to the fuel unloading machine and the inter-building coffin and required the application of considerable ingenuity and imagination. Although this equipment and the operations were developed specifically to accommodate the requirements of on-site fuel reprocessing, they were equally applicable to fuel storage and/or off-site processing. These conditions were demonstrated after the recycle of EBR-II fuel was discontinued.

TRANSFER PORT

The transfer port is a simple cylindrical valve through which the subassembly was moved. It had provisions for purging to achieve an argon gas atmosphere in conjunction with the fuel unloading machine. The port was manually operated and was normally closed. It was opened only when attached to the fuel unloading machine and appropriate procedures were performed to provide a compatible inert gas environment.

INTER-BUILDING COFFIN

The inter-building coffin was the vehicle for transporting the subassembly from the Reactor Building to the Fuel Cycle Facility. This movement was accomplished by building cranes and special transporters moving on tracks. Although the distances involved were relatively short, the path is rather complex because these operations were performed while the reactor was operating and reactor containment was required. The Reactor Building was a containment building requiring that containment be maintained at all times while the reactor was operating. To meet this requirement, the inter-building coffin was transported between buildings through a large air lock.

Figure C-29 — The inter-building coffin also included an argon gas circulating and cooling system which continued to operate at all times and at all locations as the subassembly was transported. This was accomplished by a battery-powered power supply on the inter-building coffin augmented by auxiliary power stations at appropriate locations along the travel path of the inter-building coffin.

The inter-building coffin was transferred from the fuel unloading machine loading station to a self-propelled dolly in the equipment air lock by the Reactor Building rotary bridge crane. It was transported through the length of the air lock by the dolly. It was lifted from the dolly by a crane in the Fuel Cycle Facility and transferred to another dolly stationed at floor level. Transport through the air lock was accomplished with only one hatch open at a time to maintain reactor building containment.

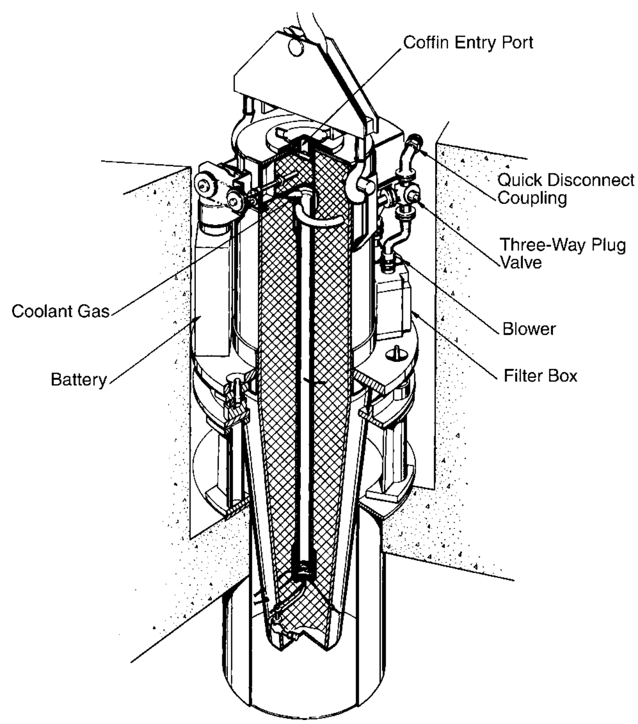

FIGURE C-29. INTER-BUILDING FUEL TRANSFER COFFIN.

In the Fuel Cycle Facility, the inter-building coffin was the vehicle in which the subassembly was cleaned of residual sodium residing on the surfaces of the subassembly and its internals, including the fuel elements. This was the radioactive primary coolant sodium from which the subassembly was removed by the fuel unloading machine. Sodium removal was accomplished by controlled oxidation followed by a final water wash. During the cleaning operation, the gas environment, which is also the forced circulation gas coolant, changed from argon to air. Cooling continued during the cleaning process.

The inter-building coffin then resumed its function as the transporter of the subassembly. The inter-building coffin, still on the floor level dolly, was lifted from the dolly and lowered by the Fuel Cycle Facility building crane to a dolly which traveled below the air cell. After the dolly was positioned below the air cell floor hatch, the subassembly was raised into the air cell by an air-cooled grapple attached to the air cell crane. At this point the transport process was complete.

The transport and transfer of reprocessed subassemblies from the Fuel Cycle Facility to the Reactor Plant followed the above stages, except for sodium removal, in reverse order. All crane, hoist, and cart operations were manually controlled and included monitoring of the supporting operations involved.

Steam Generator Design and Fabrication

A basic tenet of the EBR-II concept was to design all the parts of the plant external to the reactor and primary system as conservatively as practicable, consistent with the EBR-II mission. Part of the mission was to demonstrate electric power generation by the use of a steam-powered turbine generator, therefore, it was necessary to convert the heat produced in the reactor to steam. Since the heat generated in the reactor was transferred to liquid sodium, it was necessary to transfer that heat from sodium to water/steam to achieve this objective.

Sodium and water are totally incompatible fluids and it was necessary to ensure that they did not come in contact with each other.

A conservative approach was taken in the design of the EBR-I steam generators (i.e., triple wall concentric tubes with a copper tube between two steel tubes in a single manifold tube array). This design proved to be reliable but totally impractical for a larger plant system. The EBR-II concept incorporated the conservative feature of requiring a “double failure” before sodium and water could come in contact. This feature was incorporated into a shell and tube configuration which required considerable ingenuity in design and fabrication. The successful operation of these units attests to the wisdom of the design decisions and warrants consideration of this design for future applications.

The overall configuration of the initial design of the EBR-II individual steam generator units (the evaporator and the superheater units) is shown in Figure 3-34. The initial configuration of the total steam generator consisted of eight steam generator units and four superheater units as shown in Figure 3-33.

Details of the EBR-II evaporator (and modified superheater) design are shown in Figure 3-35. The outer tube of the duplex tube was welded to the sodium tube sheet as shown in the insert (and Figure C-30). It was a difficult weld to make because of the close spacing of the tubes and the extension of the inner tubes. An automatic welding machine was developed to accomplish it. The welding gun was placed over the inner tube extension and the weld was made between the outer tube and the sodium tube sheet in a single pass with the entire steam generator unit being rotated (one revolution).

Figure C-30 — These sectioned test welds were made prior to each unit run and calibrated by clamping the test sample on the steam generator and rotating it, as was done for the actual tube-to-tube sheet welds. The welding process was successful in manufacturing the eight steam generator units, but was not reliable for manufacture of the superheater units which employed smaller diameter tubes and closer tube spacing. Many reliable welds were made, but not consistently. The smaller tube diameter, closer tube spacing, and thinner wall were not amenable to the then available welding process.

Spare parts were available to construct two additional evaporator units. These units were modified to serve as superheaters. A decrease in steam flow area was required to simulate the smaller diameter tubes intended for the superheaters. This was accomplished by the insertion of a core tube in each tube, providing an annulus for steam flow and thus achieving the desired steam velocity through the modified superheater as shown in Figure 3-35.

The operation of these units was satisfactory. With only a slight reduction in superheated steam temperature, all efforts to successfully manufacture the original steam superheater units were terminated. It was not an objective of the EBR-II program to develop sodium-heated steam generators; it was only to generate steam reliably. This was accomplished by requiring that sodium and water/steam would not come in contact as a result of a single weld failure or a single tube failure.

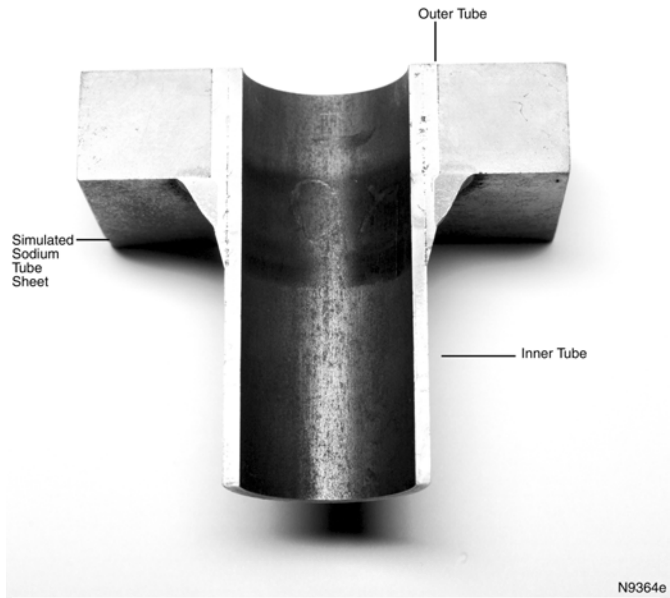

FIGURE C-30. SECTION OF TUBE-TO-TUBE SHEET WELD (TEST SAMPLE).

The key to accomplishing these objectives was the manufacture of high-quality tubes, the quality of which was verified as individual tubes and again as duplex tubes. The outer tube-to-sodium tube sheet weld has already been discussed. The inner tube to the water/steam tube sheet was a conventional butt weld. No tube leaks occurred during the 30 years of operation of EBR-II. No sodium leaks occurred at the tube-to-sodium tube sheet welds as well. One water/steam leak occurred at the tube-to-steam tube sheet. Steam leaked into the space between the two tube sheets which is open to the atmosphere. This weld was repaired (only one tube-to-tube sheet weld was involved) and the unit was returned to service and operated satisfactorily for the life of the plant. The EBR-II objective was achieved; sodium and water/steam never came in contact during the operating lifetime of the plant.

During operation, sodium flowed on the shell side and water/steam flowed in the tubes. The shell was hotter than the tubes creating a differential temperature and placing the tubes in tension and imposing stress on the tube-to-sodium tube sheet welds. It was, therefore, desirable to place the tubes in compression at room temperature, or when the entire unit was at constant temperature. Various concepts for compressing the tubes during manufacturing were considered and discarded. An alternative concept of shortening the length of the shell also was pursued. Two different methods were employed.

The first method involved heating the shell to cause it to expand in length during manufacturing. This was done after the tube sheet welds had been made at one end of the unit and prior to making the tube-to-tube sheet welds at the other end. While the shell was heated, the tubes were kept cool by placing a “stopper” in each tube near the end to be welded. Water was introduced into each tube through a manifold of smaller diameter tubes. The tube-to-tube sheet welds were then made while the tubes were water cooled and the shell electrically heated. This was a tedious and complex process because the entire unit was rotated to make the welds.

The second method involved “shortening” the shell. After the evaporator units were fabricated, a procedure was followed that shrunk the shell lengthwise and placed all of the tubes in compression and the shell in tension. Since these tubes were quite long (approximately 30 feet) they actually bent slightly in response to the compressive load. This was accomplished in the following manner. Cold water was circulated through the tubes from the inlet to outlet nozzles, while the shell was electrically heated. This caused the shell to expand in length, which was resisted by the cooled tubes in tension. In addition, the shell was heated locally around its circumference by an induction heating coil to the temperature which would cause the shell wall to yield. The compressive load on the shell caused the shell to "yield" at the local "hot ring" in the shell. The amount of power to the induction coil was carefully controlled and the amount of yield was carefully measured to ensure the correct amount of "shortening" of the shell, at which point the heating was stopped.

Both of these methods produced the desired "compression" of the tubes and low stresses in the tube-to-tube sheet welds during operation. The designers and fabricators of future units will have options available.

Another element of "EBR-II conservatism" was incorporated into the EBR-II steam system which impacted thermal performance. To minimize thermal stress in the system, resulting from temperature differential between feedwater temperature and saturation temperature at operating pressure, additional feedwater heating was used. Conventional steam extraction from the turbine was augmented by delivering steam directly from the main steam line to heater No. 4 to provide 550°F feedwater to the evaporators (saturation temperature was 580°F). In addition, constant pressure was maintained in the steam system to maintain a constant saturation temperature. Load variation was accommodated by bypassing steam directly to the condenser. The bypass system and condenser were sized for 100 percent bypass flow to provide an available heat sink for the reactor independent of the turbine generator load. These provisions, of course, reduced the thermal efficiency of the system somewhat, but achieving high thermal efficiency was not a primary objective of the EBR-II concept. On the contrary, a basic objective was the demonstration of the potential for low fuel cost which obviates the incentive for high thermal efficiency. A simplified flow diagram for the total EBR-II power cycle is shown in Figure 3-1.

APPENDIX D

APPLICABILITY OF THE EBR-II CONCEPT TO FUTURE LIQUID METAL COOLED FAST BREEDER REACTORS

Appendix D

Applicability of the EBR-II Concept to Future Liquid Metal Cooled Fast Breeder Reactor

One of the primary objectives of the EBR-II program was to establish a base for the further development of this nuclear power plant concept. The successful history of the program demonstrates that the EBR-II concept is technically feasible and operational. Still to be achieved is the extension or extrapolation of this experience to larger systems.

References were made to the expectations of the EBR-II participants and supporters that a follow-on plant would be developed which would be larger and incorporate the proven features of EBR-II, as well as improvements and features more advanced than EBR-II. Many of the participants (including the author) contemplated and speculated about an “EBR-III Concept.” Although that concept never materialized, it is appropriate to describe those features that seemed to standout and the improvements and new features that appeared to take shape during the total lifetime of EBR-II. This was an interesting and educational process for the author for more than 40 years.

This has been an on-going process, and the EBR-III concept has undergone many changes which reflect not only advancing technology, but also changes in the circumstances surrounding nuclear power. These include:

- Availability of uranium
- Availability of plutonium and the significance of breeding
- Concerns about proliferation of nuclear weapons
- “Greenhouse effect” of combustion products
- Concerns about long-term storage of spent fuel
- Concerns about shipment of spent fuel (and other radioactive materials)
- Questions related to control and cost of reprocessing.

No effort has been made to identify how each of these considerations has affected the development of an EBR-III concept, but their influence is reflected in the process. First and foremost, it is clear that the total technology does not yet exist to proceed commercially with multiple units based on available knowledge and experience. An EBR-III step is essential and it must include the appropriate provisions for additional technology development and demonstration. Second, the mission of EBR-II was not completed (i.e., the total demonstration of fuel recycling still requires development and demonstration). This must be a primary objective of EBR-III.

To formulate an EBR-III concept, it is also essential to identify key features of the EBR-II design and the incentives to retain them in a larger power station. For convenience and simplification, we will refer to this hypothetical plant as EBR-III and assume that it will be 10 to 20 times larger than EBR-II (i.e., a 200 to 400 megawatt electric liquid metal cooled fast breeder reactor power station). It should be emphasized, however, that there is no attempt to establish a preferred or optimum size for future liquid metal cooled fast breeder reactor power stations but only to establish an appropriate size for a developmental demonstration plant.

Some of the unique features of the EBR-II concept that warrant consideration and evaluation for possible inclusion in the development of an EBR-III concept include the following:

1. Size and method of support of the primary tank:

   a. There are strong incentives to minimize the size of the primary tank.

   b. There is a strong incentive to hang the tank from the top because this permits the use of a reliable second safety tank and reliability of sodium containment in the primary system.

c. The reactor size is a major factor in establishing the primary tank size.

2. Reactor cover and directed flow from the reactor to the intermediate heat exchangers:

   a. This feature, involving a moveable reactor cover, confines the hot sodium circuit and provides the necessary primary sodium system pressure to provide flow through the heat exchanger.

   b. This arrangement results in a cool primary tank since the bulk sodium in the tank is at the reactor inlet coolant temperature.

   c. This cool environment contributes to the structural integrity of the primary tank because it minimizes thermal gradients and thermal transients on the tank wall.

   d. This concept requires piping in a relatively restricted space. Leaky joints are permissible which may allow the use of floating pipes to accommodate thermal expansion.

   e. The size of the reactor cover will be an important factor in establishing component configuration in the primary tank and the size of the tank.

3. The fuel handling, transfer, and transport systems:

   a. The direct gripper with adjacent subassembly hold-down is extremely reliable and proficient in off-normal conditions. The in-vessel storage feature in conjunction with the restricted operating mode is a tremendous operational asset.

   b. Transfer from in-vessel storage out of the primary tank while the reactor is operating permits extraordinary operating flexibility.

   c. The fuel transfer and transport systems are adaptable to on-site processing or delivery offsite.

4. The shutdown cooling systems and reliability of fission product decay heat removal:

   a. One of the basic attributes of the EBR-II submerged primary system concept is fission product decay heat removal from the reactor under all circumstances by thermal convection passive circulation of sodium through the reactor.

   b. This heat is delivered to the bulk sodium in the primary tank which is a huge heat sink. It eliminates the need for rapid action.

   c. The passive systems employed in EBR-II can be augmented by active (powered) systems which do not affect their passive capability but increase their heat removal capability.

5. Fuel recycle — The Fuel Cycle Facility:

   a. The EBR-II concept was based on pyrochemical reprocessing, but the technology was not completely developed and demonstrated.

   b. Some aspects of the recycle process were satisfactorily demonstrated and are applicable to larger plants.

   c. An EBR-III concept must be capable of incorporating and demonstrating different fuel reprocessing and fabrication concepts.

Adjustments and compromises will be necessary. Some of the variables available will be identified here, but there will be no attempt to optimize them in this document. That function will be left for the designers of a real plant, but potential options that they may wish to consider will be identified. The following systems and features will be discussed for their applicability to larger reactors:

- EBR-II Reactor Concept

- EBR-II Primary System Concept
- Plant Size Considerations/Limitations
- Fuel Handling, Transfer, and Transport
- Shutdown Cooling (Fission Product Decay Heat Removal)
- Steam Generation Equipment and Cycle
- Plant Operability and Reliability
- EBR-II Fuel Cycle
- Other Concepts Considered But Not Included in EBR-II.

EBR-II REACTOR CONCEPT

The basic EBR-II reactor concept should be applicable to larger reactors. Many of the reactor features have already been incorporated in other designs. These include the close-packed geometry of subassemblies located and supported in a lower inlet plenum-grid structure. In a larger reactor it should be advantageous to use larger subassemblies and a much smaller proportion of blanket subassemblies. EBR-II was designed to demonstrate the long-term potential of the liquid metal cooled fast breeder reactor for electric power generation on the uranium-238 fuel cycle, with plutonium being the catalyst in the process. Part of that demonstration included enhancing the breeding capability to ensure the availability of plutonium to fuel additional plants required to accommodate the growth in estimated power demand. The near-term objectives, requirements, and emphasis are now much different. Plutonium is now a glut, but not a waste product, while other actinides and fission products are a difficult real waste product. Emphasis is now on consuming plutonium (and the other actinides). Therefore, for some time to come, the need for breeding or conversion of uranium-238 into plutonium will be determined primarily by reactor performance and fuel cycle requirements. It may be desired to achieve a breeding ratio of about unity to operate on a pure uranium-238 cycle, or to achieve only sufficient uranium to plutonium conversion to control reactivity as burn up proceeds. It is likely that only a relatively thin blanket will be needed.

This possibility introduces an option that could significantly reduce the demands on the fuel handling system and the diameter of the rotating plugs. Such a reactor configuration should provide the potential for a proportionately simpler and smaller system. The core could be surrounded by a relatively thin blanket, comparable to the EBR-II inner blanket, which in turn could be surrounded by a reflector region. The core and blanket subassemblies could incorporate the basic EBR-II concept of subassembly design, grid/plenum design, coolant supply, and orificing.

The configuration of the blanket and/or reflector obviously should result from the desired operating characteristics such as, breeding, reactivity change, actinide burning, materials irradiation, etc. Potential uses of liquid metal cooled fast breeder reactors should not be restricted; they have a tremendous long-term potential not only for electric power generation, but for high-level waste disposal and for possible material production requiring a high, fast neutron flux. The reactor size need not increase proportionately to the increase in power level over EBR-II.

The larger reactor employing larger subassemblies should easily incorporate the EBR-II divided coolant system, utilizing a high-pressure supply for the subassemblies with high heat generation and low pressure for low heat generation subassemblies. Similarly, the hydraulic hold-down feature incorporated into the EBR-II fuel and inner blanket subassemblies has much merit and should be applicable to EBR-III. The EBR-II concept of variable coolant flow orificing, established by subassembly location in the reactor, should also be applicable and much more effective in a larger reactor.

Vertical movement of control rods by drives located above the reactor is a feature that most logically extends to larger reactors. It should be easier with greater space available because the control rods will be farther apart. EBR-II has demonstrated that movement of fuel, as well as a combination of fuel and absorber, is feasible and should be applicable. The large plant designers have options, particularly if it is not necessary to maximize breeding gain.

The detailed design of the subassemblies, control rods, and other reactor components must be integrated into the fuel handling and fuel transfer systems, as well as the reactor configuration. The

EBR-II concept of using a common design for the handles and attachments worked well and should be applicable to future plants.

EBR-II PRIMARY SYSTEM CONCEPT

The merits of the EBR-II submerged primary system concept were recognized quickly in France, Great Britain, and the Soviet Union. The general EBR-II concept was adopted, but with some significant exceptions. One was the method of support of the primary tank. Another was the directed flow concept of EBR-II using a cover over the reactor and sodium piping from the reactor to the intermediate heat exchanger. These features have some specific advantages and warrant discussion as to their applicability to larger plants. In addition, a variety of fuel handling concepts were employed. The future large plant designer will have many options.

The EBR-II primary tank was hung to maximize the reliability of containment of the sodium in the primary tank. Reliability of containment of the primary sodium and assurance that the reactor would be submerged in sodium was central to the EBR-II concept of reliability and public safety. Hanging the tank permitted the use of a safety tank surrounding the primary tank, whose only loading would be sodium if the primary tank leaked. All designs using bottom support of the primary tank involve support structures that penetrate the safety tank and introduce potential points of stress and leakage possibly resulting in a common mode failure. Bottom support also complicates thermal expansion provisions.

The EBR-II concept may be more sensitive to primary tank size than some other submerged primary system concepts. The ability to hang the tank is a major consideration. The designers of EBR-II recognized the thermal expansion requirements and selected a hung system, which also provided a flexible support system. By a process of evolution, a double pin hanger was superceded by a roller design. A more sophisticated and complex system may be required for a large system, but there are many options including semi-fixed supports, prestressed to operating condition geometry.

The EBR-II primary tank and contents temperature were maintained for approximately 40 years, beginning with the initial filling with sodium. Most of that time the system was maintained at the basic operating temperature of 700°F. Significant temperature changes were infrequent and extremely slow. This experience would suggest that a primary tank for a larger system could be hung from large straps, which would bow as the tank expands and contracts, and which could be positioned and fixed at a diameter equivalent to that which would exist at some intermediate temperature such as 400°F–500°F. These straps could be rectangular in cross section to bend in the radial tank expanding direction, but rigid at 90 degrees to the expansion direction. It would appear that a much larger tank can be hung and maintain accurate positioning while accommodating thermal expansion. The coolant retaining capability of the primary tank is a fundamental reliability and safety asset of the EBR-II concept that should be extended to larger units, if at all feasible.

Similarly, the advantages of the “cool boundary-closed hot leg” primary system concept incorporated into EBR-II, contributes to the capability of the system to minimize the base temperature of the primary system affecting thermal expansion and temperature gradients and accommodates abnormal operation events.

The directed flow concept used in EBR-II presents a more complex challenge for EBR-III (or other larger systems). As discussed earlier, the size of the reactor cover needed to close the system and provide pipe-directed sodium flow to the heat exchangers is a primary factor in establishing feasibility. A second factor is the space requirements for the hot leg piping from the reactor to the heat exchangers. Expansion of the primary tank and thermal movement of the outlet piping both affect the piping loads.

In EBR-II, expansion was accommodated by generous expansion bends in the piping from the reactor to the intermediate heat exchanger. This required considerable space, but was permissible in EBR-II because only one pipe (and one heat exchanger) were required. The outlet piping and design were further complicated by the desire to ensure natural convection circulation of the coolant through the reactor under all possible operating conditions, including loss of heat removal by the secondary system. The EBR-II concept of piping and intermediate heat exchanger configuration, auxiliary pump, etc. are not suitable for EBR-III. To accommodate both requirements, the piping and intermediate heat exchanger concepts for power operation will be discussed first, and the provisions for passive fission product heat removal will be discussed later.

In a large system, multiple units, probably three or four outlet pipe and intermediate heat exchanger circuits, will be required. The piping involves a horizontal outlet from the reactor, a vertical run, and a horizontal inlet into the intermediate heat exchanger. Since this piping is submerged in the primary sodium, it need not be fixed and leak tight. Use of floating pipes with the capability to move at the connection points (and leak), could simplify the piping design immensely and reduce the space required for the pipes. Figure D-1 depicts a simplified arrangement for such a piping system between the reactor and intermediate heat exchanger. This hot leg piping would require some insulation. EBR-II employed a double-walled pipe with static sodium between the two pipes. It need not be highly efficient insulation since the heat is not lost from the system, but does increase the work (pumping power) required to convert the energy produced in the reactor to steam.

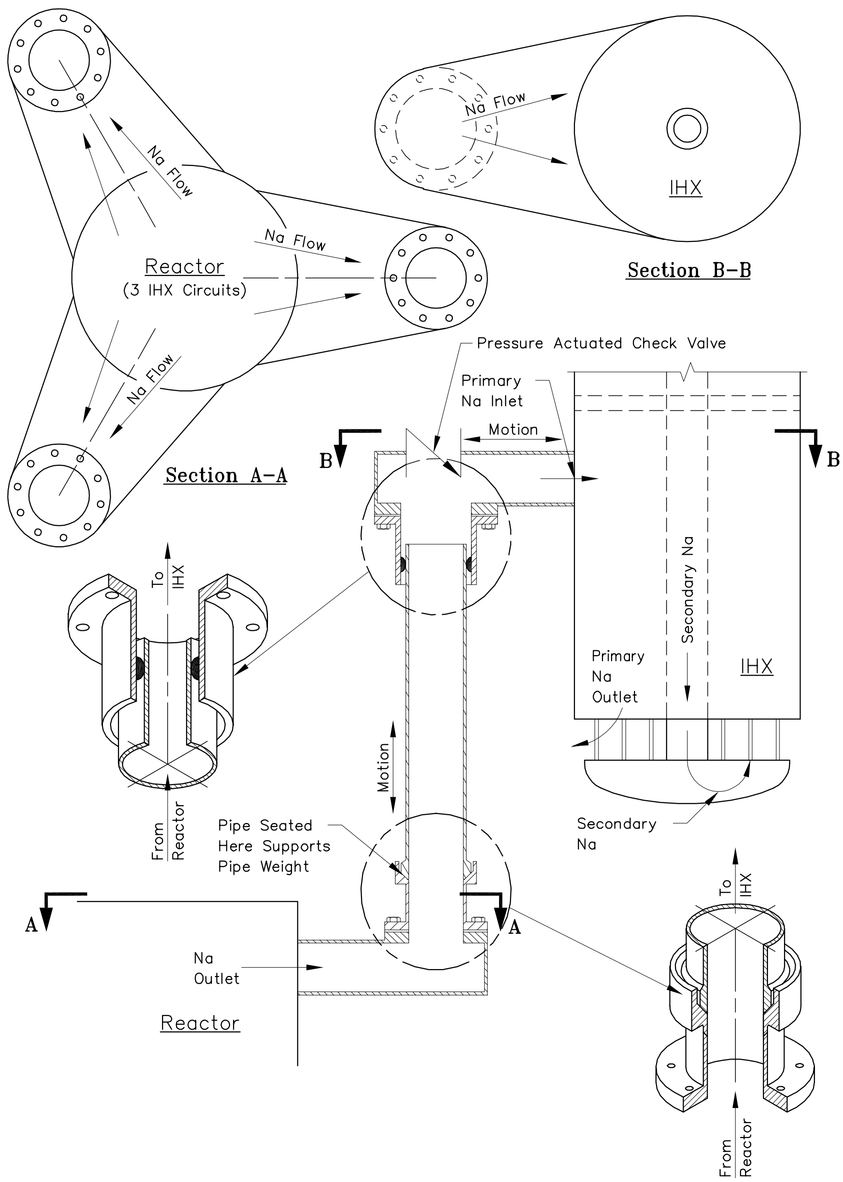

FIGURE D-1. REACTOR IHX PIPING.

It should be possible to use the weight of the pipe to affect a seat and seal at the lower connection at the reactor outlet, similar to the arrangement employed at the EBR-II pump to pipe connection. The upper connection at the intermediate heat exchanger can be a straight slip joint which can accommodate pipe movement vertically and horizontally. It should be noted that the operating sodium pressure at both of these connections is relatively low, as compared to the EBR-II pump connection (i.e., the pressure drop through the shell side of the intermediate heat exchanger).

The other feature of the EBR-II directed hot leg flow system involves the reactor cover. A larger reactor will inevitably require a larger cover, but not proportionately larger than EBR-II. As discussed above, the reactor diameter can be reduced by minimizing radial blanket thickness. Also, experience did not indicate that the relatively large cover used in EBR-II was a problem. The EBR-II cover is raised and supported by two shafts; three might be more appropriate for a larger cover (and three points establish a plane). In the lowered position, the cover rests on the reactor vessel, and in EBR-II, is clamped to the vessel by three clamps to resist the coolant pressure in the reactor upper plenum. The cover raising/lowering drives are mounted on the small rotating plug. Since this action only occurs at the reactor operating position, the drives could be located elsewhere and brought into the drive position at this location. Since a raised position is required at all times that the rotating plugs are in the fuel handling mode, the cover could be locked in the raised position and disconnected from the drives. Also, the need for clamps can be avoided if a three shaft raising/lowering system is used. The cover can be held down, if necessary, by weights or hydraulics. This approach could reduce the clutter on top of the small rotating plug, and remove the necessity for penetrations to accommodate the cover clamp drives. The EBR-II feature of engaging the cover to the small rotating plug with pins to avoid a swinging load during plug rotation warrants consideration.

Although the primary function of the reactor cover is to enclose the outlet coolant plenum, it also provides a vehicle for guiding the control rod drives close to the reactor. This proved to be a major asset in the EBR-II design because the control rod spacing was quite small, which limited the size of the drives resulting in rather long, small-diameter drive shafts. These shafts incorporated a guide bearing in the cover where each shaft passed through a precisely located guide sleeve. This guide support feature was essential in the EBR-II design.

Another consideration related to the primary sodium cooling system involves the EBR-II requirement that fission product decay heat be removed passively from the fuel at all times and under all conditions. This was accomplished by natural circulation of sodium through the subassemblies; when they were in the reactor, or when the subassemblies were out of the reactor during fuel handling and storage. The requirement that this heat removal would be achieved in the reactor impacted the design of the coolant circuit including the reactor, the coolant piping, and the heat exchanger. The basic concept was simple; remove the fission product decay heat from the fuel in the subassembly and transfer it to the bulk sodium in the primary tank under all conceivable reactor shutdown conditions. (Removal of this heat from the bulk sodium is discussed in the “Shutdown Cooling” section.)

The impact of this requirement for heat removal from the fuel on the design of the EBR-II primary sodium coolant system has been described with particular emphasis on the auxiliary pump and the arrangement of the intermediate heat exchanger and the primary system piping to ensure coolant flow.

It may be possible to modify the EBR-II concept to achieve the same results with fewer restrictions. The EBR-II passive heat removal concept functioned just as well with the cover raised as it did with the cover closed. With the cover raised, the natural convection flow was straight up from the reactor directly into the bulk volume of sodium in the primary tank. This would suggest that removal of fission product heat can be achieved in the same manner if the cover is down or up (i.e., if the upper coolant plenum above the reactor exists, and is open to the primary tank sodium). This permits natural convection flow through the cover or through the sodium outlet pipes when the reactor is shut down, and can be achieved by check valves activated by coolant pressure. There is an additional positive and reliable system to provide a desired coolant flow path through the cover: the control rod drives. The drives pass through sleeves in the cover. The sleeves provided a guide for the drives close to the reactor and controlled leakage of coolant through the cover to acceptable levels.

The control rods and drives are primarily in the up position when the reactor is operating and the primary coolant system is operating under forced flow conditions. They are in the down position when the reactor is shut down. They are placed in this down position quickly and reliably when the reactor is scrammed. A combination bearing labyrinth seal and flow control system can be incorporated into the control rod drives, which will reliably open the cover to permit coolant flow through the cover when the reactor is shut down; quickly or slowly, but reliably. Figure D-2 is a drawing of a modification of the EBR-II control rod drive design which would provide this feature. It involves a modification in the design of the bearing guide on the control rod drive shaft that moves in the guide sleeve in the cover as the control rod drive provides vertical motion to the control rod. The lower section of the bearing guide contains circumferential grooves that act as a labyrinth seal (similar to the design which was incorporated into EBR-II), while the upper section contains vertical longitudinal grooves that permit sodium flow between the drive shaft and the guide sleeve.

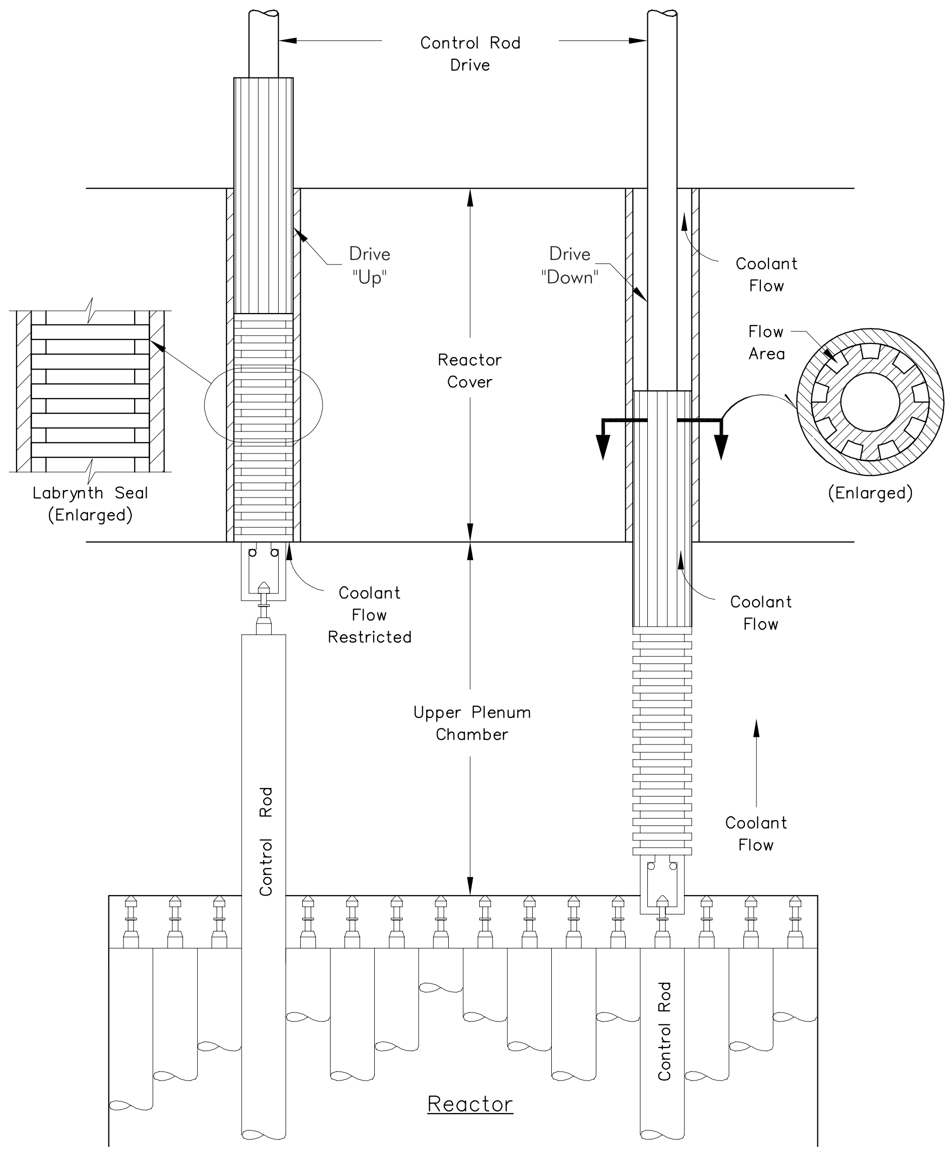

FIGURE D-2. EBR-II CONTROL ROD DRIVE DESIGN.

A second, completely independent, natural convection coolant flow path can be provided by appropriate check valves. A convenient location for such a check valve would be at the inlet nozzle to the intermediate heat exchangers as indicated in Figure D-1. These check valves would be operated by coolant pressure and would open when forced convection flow ceased. This system would provide an additional natural convection flow path for the coolant and transfer of fission product decay heat from the fuel to the bulk sodium in the primary tank.

With assurance that natural convection coolant flow will occur whenever the reactor is shut down, and fission product decay heat will be transferred to the bulk sodium in the primary tank, the total coolant flow system design can be simplified. The piping and the intermediate heat exchanger can be designed for power operation only, with forced circulation of the sodium coolant through the system. The heat exchanger will have no natural convection flow requirements. An auxiliary pump will not be required. Fission product decay heat removal will be reliable. The EBR-II concept will be preserved, but accomplished differently. Only the primary sodium cooling system will be involved in removing fission product decay heat from the fuel at any and all locations within the primary tank. The secondary sodium system and all of the power-related systems involved in energy transfer and electricity generation can be nonsafety-related, commercial grade. Only the components and systems in the primary system and the shutdown cooling system will be nuclear safety grade. (The shutdown cooling system that removes heat from the bulk primary sodium, will be discussed later in this appendix.)

The system requirements for fission product decay heat removal shutdown cooling for EBR-III will be easier than those required for pressurized systems, because of the time available for heat removal from the system to begin functioning. Once the fission product decay heat is transferred to the bulk sodium in the primary tank, considerable time is available to remove this heat because of the heat capacity of this large volume of sodium.

The elimination of a shutdown cooling requirement for the intermediate heat exchanger should permit significant latitude in design of the piping and heat exchangers in a larger reactor system. This capability should further enhance the applicability of the EBR-II directed hot leg coolant flow concept.

Finally, it should be emphasized that the use of the control rod drives to create a desired coolant flow path through the cover should not be permitted to alter the guidance of the drives through the cover. This is an extremely important function performed by the guide sleeves through the cover and contributes to the reliability and accuracy of the control rod drive system. Control rod guidance can be achieved while also providing the coolant leakage control and the coolant flow control required to achieve the desired operational characteristics.

The pump and inlet sodium piping designs incorporated in EBR-II should be directly applicable to larger units. The piping is at ambient temperature and the piping connection between the pump and the inlet piping, required to permit pump removal, is flexible. Pump removal was demonstrated, as well as the permissibility of a leaky pump-to-piping connection.

It would appear that these unique features of the EBR-II primary system concept can be extended to much larger plants. If so, they should be given serious consideration to avoid some of the problems which have been encountered in other submerged reactor concepts.

PLANT SIZE CONSIDERATIONS/LIMITATIONS

As discussed in the previous section, EBR-II design features incorporated into the reactor and primary sodium system introduce design challenges of scale. There does not appear to be size limitations outside the primary system, and larger reactors have been built in other countries. This latitude in the size of the balance of plant results from the fact that increased capacity can be achieved by adding additional systems and these systems are not space limited. Therefore, it can be concluded that the applicability of EBR-II technology to larger plants will be controlled by the primary system.

FUEL HANDLING, TRANSFER, AND TRANSPORT

The fuel handling, transfer, and transport systems were described earlier. They have been effective, reliable, flexible, and warrant serious consideration for application to EBR-III.

Direct positive gripping of the subassembly is an important attribute because this process provides no visual feedback. In combination with the hold-down of adjacent subassemblies, these features provide the capability to accommodate abnormal conditions and requirements. This capability is extremely important in the reactor core region requiring the handling of fuel subassemblies. As the burnup and residence times increase, the potential for abnormal conditions to arise will increase, including swelling and distortion of the subassemblies. This can also occur as the reactor ages. As described earlier, EBR-II operated for a much longer time than originally contemplated. This was made possible, at least in part, by the hold-down feature. Many of the subassemblies were difficult to remove and without the hold-down feature, some operational problems would have occurred.

It may be possible to reduce the size of the rotating plugs by using a second “reflector fuel handling” system to service the outer perimeter reflector subassemblies in the reactor. Demands on this system may be reduced if the requirements for handling these subassemblies are less severe (i.e., no fuel, lower neutron flux, etc.). If so, experience gained at other facilities also may be applicable. The fuel handling equipment and system at the Fermi I liquid metal cooled fast breeder reactor and other liquid metal cooled fast breeder reactor should be reviewed and evaluated.

For example, the Fermi I Fast Reactor employed an off-set handling mechanism (and a single rotating plug) to handle subassemblies. This arrangement did not control the angular position of the subassembly during fuel handling. The subassembly was rotated by guide shoes on the subassembly as it was lowered into the appropriate position in the reactor. This same concept could be applied to the reflector subassemblies at the outer periphery of the reactor. To simplify the process even further, these subassemblies could be circular in cross section, the same diameter as the hexagonal subassemblies dimension across flats as shown in Figure D-3 and Figure D-4. These reflector subassemblies would require no features to identify or control their angular orientation, but could be handled by the same components used in handling and transfer operations of fuel and blanket subassemblies.

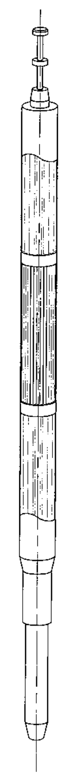

FIGURE D-3. REFLECTOR ASSEMBLY.

The reflector subassemblies, which would also engage the transfer arm, could be circular rather than rectangular as shown in Figure D-4. It would contain no angular orientation slots at the top or bottom (of the subassembly). The fuel handling system and the reflector handling system would use the transfer arm and the remainder of the fuel transfer and transport systems as appropriate. With this dual system, the reflector handling system would be used very little. It would permit a smaller diameter rotating plug system, and thus allow more space on the top of the primary tank for other equipment.

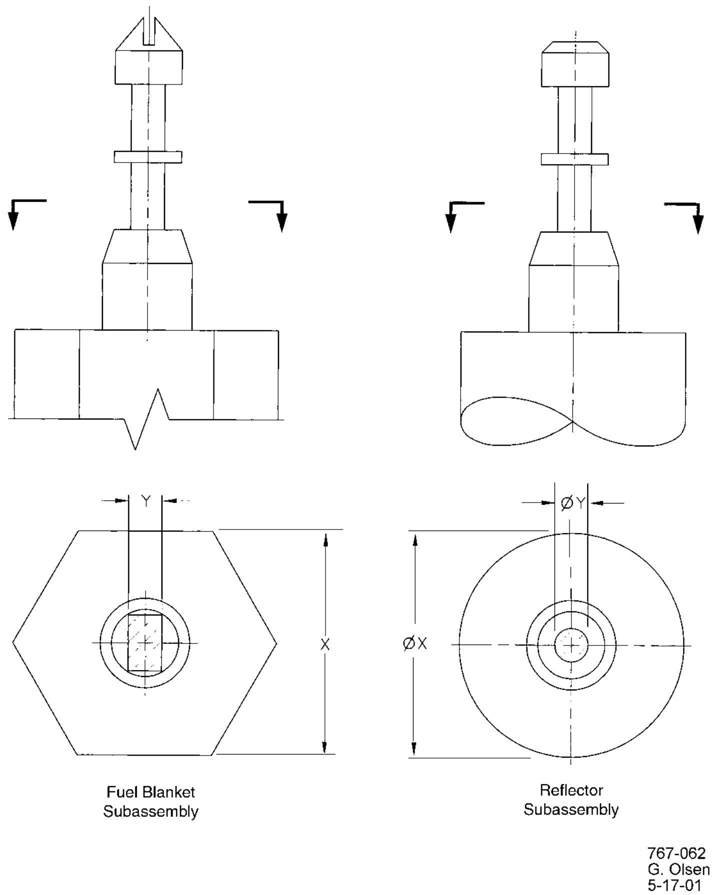

FIGURE D-4. SUBASSEMBLY UPPER ADAPTER.

The use of intermediate storage of subassemblies in the primary tank, submerged in sodium, is central to the EBR-II fuel handling/transfer concept. It permits and provides several important attributes to these processes including:

- Rapid exchange of subassemblies in the refueling process
- Storage with reliable, effective cooling to remove fission product decay heat
- Storage of new fuel permitting quick access for subassembly exchange in the refueling process
- Short duration reactor shut down for refueling
- Complete flexibility in fuel transfer and transport to accommodate external requirements, independent of reactor operation.

All of these attributes will be important to the success of a larger plant and should be incorporated into the design wherever possible.

There should be significant opportunities to improve upon the details of the specific design of components, but the basic principles should be applicable. Although the subassembly for a large reactor will certainly be larger than those employed in EBR-II, there does not appear to be any size-related limitations on the EBR-II concept. These systems functioned admirably while incorporating the technology of the 1950s. It is a sound concept.

Although the use of an intermediate storage capability was incorporated primarily to be compatible with on-site fuel recycling, it is equally applicable to off-site recycling to accommodate shipping schedules, heat removal requirements, etc. Also, shorter reactor refueling shutdowns are a significant asset, even though much higher permissible fuel burnup and less frequent refueling schedules than were anticipated for EBR-II will prevail in the future.

It was necessary for the EBR-II fuel transfer/transport systems to accommodate a unique requirement produced by the EBR-II fuel cycle; the recycled subassemblies required forced convection gas cooling to remove fission product decay heat. Therefore, the EBR-II concept included rapid return of each recycled fuel subassembly to sodium cooling in the storage rack in the primary tank after manufacture, to minimize the duration of gas cooling. Although this capability is only needed if the recycled product requires cooling, it is a convenient process for all new fuel subassemblies. It is advantageous to have the reload fuel stored at temperature and in the proper environment, ready to be installed directly in the reactor. Since the reload fuel can be transferred into the storage facility while the reactor is operating, the time involved is not critical. This can be an important consideration because this transfer involves the change of the environment from air to inert gas to sodium, and although time consuming, it's relatively unimportant if performed while the reactor is operating. Therefore, this process can be applicable even if the specific fuel cycle requirements of EBR-II do not apply.

Fuel handling, transfer, and transport of EBR-II have shown that the concept is sound, versatile, and reliable. However, improvements can be made to the methodology and implementation. For example, the process involves the definition of each specific subassembly position in the reactor and the storage rack. In addition, each of these subassembly positions incorporates a subassembly angular position. All of this information is translated into angular positions of the gripper, the rotating plugs, the storage rack and, to a lesser extent, the transfer arm. The key identifiers are the subassembly location in the reactor and in the storage rack. In EBR-II, this consisted of 637 discrete positions in the reactor and 75 discrete positions in the storage rack. As described earlier, EBR-II employed a punch card system to translate this position information to the required rotational and angular information needed to properly direct the equipment involved. This represented application of 1950s technology.

The technology of the 21st century should make implementation of the EBR-III concept far easier, and much more accurate. Also, methods of verification of actions and records of actions should be far superior to those of EBR-II.

One of the operations involved in the EBR-II fuel handling and transfer process susceptible to significant improvement is the transfer arm. It was recognized that the operations performed by the transfer arm were critical to the total EBR-II concept. These were the operations for which a visible confirmation would be most desirable, but that direct vision could not be provided. At the time, it was concluded that a series of manual operations, with feedback provided by "feel" to the person performing the operations, was the most reliable concept available. To enhance the reliability of this concept, the operations were subdivided into simple activities that were easily visualized and confirmed by feel of the hands of the operator.

Tremendous advances have been made in the technology of controlled positioning and feedback of position indications. For example, automatic welding of complex components, and manufacture of microchips. It would appear that a transfer arm-type operation need not be performed manually. The entire fuel handling, transfer, and parts of the transport process are amenable to computer control. It may be beneficial to develop such a system for the EBR-III concept. It would appear that the future designers will be able to retain the EBR-II basic concept and improve upon it. For example, it might be advantageous to provide additional vertical travel capability to the transfer arm and require only rotation of the storage basket. That option, combined with the experience gained at other liquid metal cooled fast breeder reactor facilities, should make this part of future liquid metal cooled fast breeder reactor operations even more successful and reliable.

Finally, the EBR-II fuel recycle concept involved an evolution that began with a disassembly cell located immediately above the primary tank (as described earlier), that later became the "air cell" in the Fuel Cycle Facility. This evolutionary process reflected the selection of a specific fuel reprocessing and recycling concept for EBR-II. This was a bold decision, and although it was quite successful, the total demonstration was not completed. A complete closed fuel cycle for EBR-II was not developed. Therefore, a critical requirement for EBR-III must include the capability to contribute to the development of closed fuel cycles and to demonstrate operation with recycled fuel.

To accomplish these objectives, EBR-III should include facilities similar to those incorporated in the EBR-II Fuel Cycle Facility air cell. This would include facilities and equipment to perform the following functions:

A. Receive a spent subassembly and remove the adhering sodium coolant.

B. Disassemble the subassembly and package its component parts (particularly the fuel elements).

C. Transfer all of the components from the facility appropriately.

D. Receive all of the component parts of a subassembly (as well as complete subassemblies).

E. Assemble a subassembly.

F. Transfer a subassembly to the reactor.

G. Provide fission product decay heat removal as required.

It is not recommended that this facility be located above the reactor as first contemplated for EBR-II. However, consideration should be given to locating it in the Reactor Building to minimize and simplify the operations involved in the transport of a spent fuel assembly. A logical location would be approximately where an EBR-II subassembly is transferred between the fuel unloading machine and the inter-building coffin. A possible location for the disassembly/assembly cell is shown in Figure D-5. A subassembly transfer machine (similar to the EBR-II fuel unloading machine) would transfer subassemblies between the primary tank and the subassembly cell. At each location the transfer would be between the machine and a location below the machine. The movements involved in the transfers would be quite similar but the work environments would be very different; sodium in the primary tank and air in the subassembly cell. The transition between these two environments would occur in the transfer machine, just as it did very successfully in the EBR-II fuel unloading machine for more than 30 years.

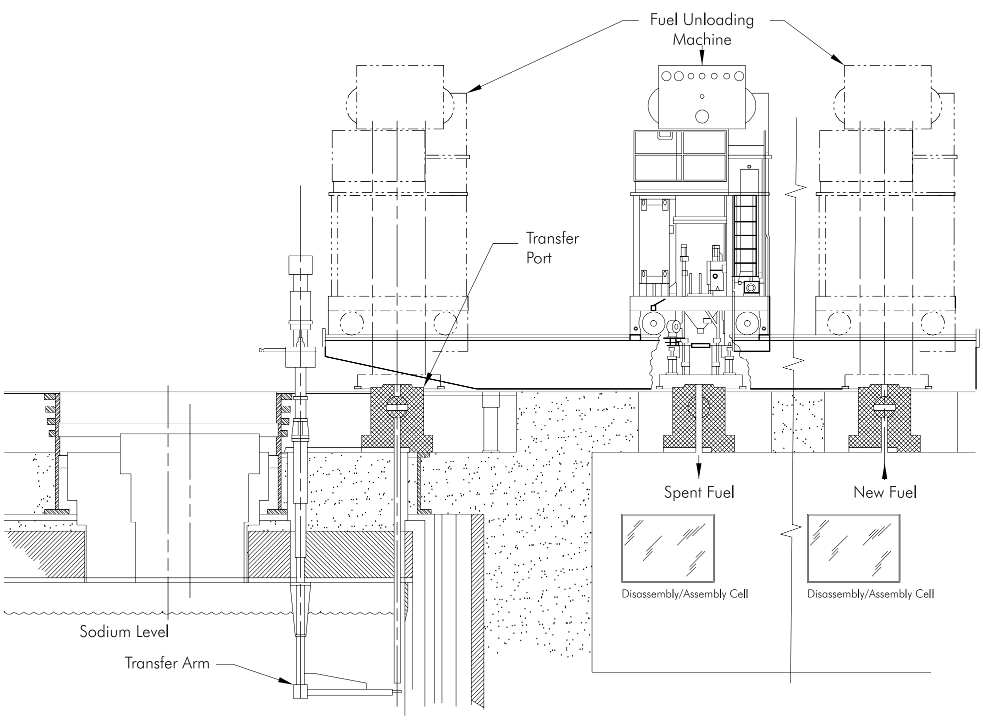

FIGURE D-5. POSSIBLE LOCATION FOR THE DISASSEMBLY/ASSEMBLY CELL.

Similarly, the operations to be performed in the subassembly cell would be essentially the same as those performed in the EBR-II inter-building coffin, and Fuel Cycle Facility air cell; namely, remove sodium coolant from a spent subassembly, disassemble a fuel subassembly, and package fuel elements for transfer. Recycled fuel elements would be assembled into subassemblies for return to the reactor.

This arrangement would provide the flexibility to develop and demonstrate the feasibility and acceptability of various processes for recycling the fuel. Recycled fuel elements would be the product delivered to EBR-III. They could be recycled onsite or offsite. The plant could accept either. The balance of parts comprising the complete subassembly could be supplied as most convenient. The primary emphasis could be given to fuel recycle. Blanket reprocessing and breeding could be deferred as long as plutonium and uranium-238 continue to be available. Demonstration of transmutation and actinide utilization will be of greater initial interest.

This aspect of an EBR-III concept emphasizes the importance of fuel recycle. It was prevalent during the development of the EBR-II concept and was incorporated into the design and initial operating plan. Much of the EBR-II experience is applicable to the continuation and extension of this strategy.

This aspect of developing an EBR-III concept is probably the most important consideration in making fast power reactors available for future commercial use. To achieve the potential of this technology requires the fuel to be recycled until it is “gone” (i.e., until it has been transmuted completely). Complete fuel recycle must be demonstrated on a continuing basis with the reactor operating with equilibrium fuel composition.

SHUTDOWN COOLING (FISSION PRODUCT DECAY HEAT REMOVAL)

EBR-II incorporates a passive cooling system for removal of fission product decay heat, which requires no power driven equipment and which functions entirely by natural convection of the coolant systems. It is in continuous operation, at a low heat removal rate, and becomes fully operational automatically in response to a fail-safe signal.

It would appear that this basic concept could be extended to much larger liquid metal cooled fast breeder reactor power plant systems by incorporating additional features. One possibility would be to use a similar system, but incorporate two levels of cooling capability. The base system would be a passive system similar to EBR-II and would be capable of providing sufficient cooling to ensure that fuel melting or significant fuel damage would not occur and public safety would not be jeopardized. This system could be augmented by emergency powered components that would provide additional heat removal capability by the same system and maintain normal temperatures in the system. This could be achieved by adding forced convection capability to the normal natural convection systems.

The EBR-II shutdown cooling system lends itself to such augmented cooling capability. The sodium potassium lines between the shut-down coolers in the primary tank and the sodium potassium to air heat exchangers in the cooling stacks could be provided with direct current electromagnetic pumps driven by rectifier/battery power supplies which are extremely reliable. Natural circulation of air in the cooling stacks could be augmented by emergency power driven fans in the stacks. These provisions could increase the heat removal capability of the passive system significantly.

These would be simple systems as compared to those required by light water reactor systems, or any pressurized system where loss of coolant is a primary concern. The normal operation of emergency power systems would ensure reliable removal of fission product decay heat. Even in the improbable event of failure of these emergency systems to operate, the passive heat removal capability of the system would ensure that significant fuel damage would not occur and the consequences would be only economic; there would be no public hazard.

As a result, the normal emergency response of the shutdown cooling system could use emergency power supplies to provide the required heat removal. The heat capacity of the bulk primary sodium provides a considerable amount of time for these powered systems to operate. In EBR-II many hours are available, but even in a larger plant, considerable time would be available for corrective action. Emergency response in the short time required by pressurized systems would not be required. Much of this reliability of shutdown cooling results because this type of plant operates at essentially atmospheric pressure and the reactor core is submerged in the coolant which has unrestricted movement and is essentially unlimited. Keeping the core covered by coolant and ensuring unrestricted flow of coolant is inherent in the basic design.

The reliability features such as valveless piping of the sodium potassium coolant system and normally open shutters on the air stack, held closed by electromagnets, can be retained even though the system is dual powered. The EBR-II system design can be applied directly to the large plant design and augmented by direct current electromagnetic pumps on the sodium potassium piping and emergency powered circulating fans in the air stacks.

The EBR-II concept incorporates the capability of providing a level of reliability of fission product decay heat removal superior to any other nuclear reactor power system.

STEAM GENERATION EQUIPMENT AND CYCLE

The EBR-II concept incorporated a conservative approach to the steam cycle and steam generation equipment. This approach resulted from two major considerations or conclusions.

First, a significant factor in considering the liquid metal cooled fast breeder reactor power cycle fueled by uranium-238 is the potential for low fuel cost. Under these conditions there is little incentive to achieve extremely high thermal efficiency and more incentive to achieve low plant capital cost and high plant capacity factor. This philosophy results in modest state-of-the-art steam-cycle conditions using proven reliable technology.

Second, the objective of EBR-II was to establish and demonstrate the technical feasibility and utility of the liquid metal cooled fast breeder reactor as an energy source for the production of electricity. To accomplish this objective, it was prudent to minimize the probability of plant shutdowns and nonoperation resulting from steam system problems.

The relatively conservative steam cycle conditions selected for EBR-II were reasonably standard for small fossil fueled plants in the 1950s. In addition, these standard conditions were made even more conservative by additional feedwater heating to minimize thermal transients. The use of recirculation steam generation units with a large steam drum added further stability to the steam cycle. Temperature transients and local thermal stresses, as can be produced in once-through and other systems, are minimized.

The EBR-II concept is perhaps even more conservative with respect to the design of the steam generation equipment (i.e., the steam generators and superheaters). The design of these double tube sheet units have been described earlier. This conservatism resulted from the strong incentives to avoid any contact between these incompatible fluids, water or steam and sodium.

The successful lifetime operation of this system and the absence of failures or problems (contrasted to experience at other liquid metal cooled fast breeder reactor facilities) would suggest that a conservative approach to the design of these systems and components is warranted.

A conservative steam cycle can certainly be applied. Special features, such as additional feedwater heating to minimize thermal gradients or shock can be considered. A more difficult evolution involves the design of the EBR-II steam generators and superheaters. Much larger units will be required. Insufficient work has been done to establish their size limitations. Experience with pressurized water reactor steam generators does not produce confidence that absolute leak tightness for a plant lifetime is achievable. It is well established that sodium is an easier fluid to handle on the shell side of a tube and shell exchanger than is water (e.g., crevices can be tolerated and low pressure on the shell side). But it also is well established that even a small leak of water or steam into sodium will require suspension of operation. Therefore, the EBR-II experience should be considered and applied as extensively as possible.

A development program to extend EBR-II design features to larger size should be undertaken. One of the major features incorporated into the EBR-II steam generator design has not been given adequate consideration. Shell and tube steam generators naturally employ sodium in the shell side at low pressure. Since this is the hot side, the tubes are in tension, which causes stress at the tube-to-tube sheet weld. “Hockey Stick,” “U-Tube,” and other arrangements have been advanced to ameliorate this condition. The EBR-II solution of placing the tubes in compression, actually bending because of their length, is a far simpler solution which permits the use of straight tubes. The use of straight tubes permits the use of duplex tubes, which are not amenable to bending. The use of duplex tubes enhances the reliability of the tubes tremendously. The probability of both tubes leaking at the same location is reduced enormously.

Also, the EBR-II approach of shrinking the length of the shell and (slightly) bending the tubes provides much latitude to the designer. The stress in the tube-to-tube sheet weld is adjustable. It can be made essentially zero by maintaining the tube slightly bent at operating conditions by shrinking the length of the shell appropriately. There are a variety of methods for effectively shrinking the shell. EBR-II employed two; there are certainly others. The most obvious is to cut the shell after assembly, remove the desired shortening band, and rewelding. Or conversely, the unit can be constructed with a gap in the shell bolted mechanically, and welded after the tubes are installed.

In a simple shell and tube heat exchanger, there are two locations where water and/or steam are separated from the sodium by a barrier: the tubes and the tube sheet. Of these, the tube is by far the most critical barrier because there is so much more area involved, while the tube sheet represents a much more imposing barrier. Therefore, the use of duplex tubes should be considered the most important feature of the EBR-II steam generator concept.

The use of double tube sheets created manufacturing difficulties in the EBR-II steam generators and the original design of superheaters could not be manufactured reliably. Since the duplex tube feature is much more critical than the double tube sheet, a duplex tube with a single tube sheet concept should warrant consideration. It is well established that even a small leak in a tube will quickly escalate to a large failure. Many tests have been conducted and have demonstrated that a small leak cannot be tolerated. This can be compared to pressurized water reactor experience which has demonstrated that many leaks can be tolerated and the leakage is limited by the amount of primary radioactive water that can be tolerated in the steam system.

The use of double tubes and single tube sheets would simplify the EBR-II concept and could be enhanced by restricting the flow area between the tube and tube sheet to restrict water or steam leakage into the sodium in the event of a weld failure. The objective should be to create the necessary conditions at the tube-to-tube sheet connection that in the event of a leak in the weld, the leak between the tube and the tube sheet will be small enough and slow enough to permit detection, response and correction before any significant damage is caused. This can not be achieved if a leak occurs in a tube within the steam generation unit.

This objective would suggest that some imaginative designs be developed for the tube-to-tube sheet connection. There already exists an inherent, built in advantage by the nature of the thick tube sheet required and the resultant long channel between the tube outside diameter and the hole in the tube sheet. This channel should be made extremely small. It can be made negative at room temperature. Rolling the tubes in the tube sheet is a well-established practice, but it may not be sufficient.

Perhaps a more exotic approach should be considered. A shrink fit could be made. The ends of the duplex tubes could be machined to a close tolerance in diameter. The holes in the tube sheet could be bored to a close tolerance, and the combination could provide an interference fit. Assembly would be achieved by heating the tube sheet and cooling the tubes. Rather extreme conditions could be established if necessary to achieve the desired fit. For example, the tube sheet could be heated slowly to the desired temperature. Because of its mass, it would not cool quickly. The tubes could be cooled with dry ice, packed into the tubes to provide appropriate clearance for assembly. The amount of shrink fit could be established to provide a leak-tight fit to back up the tube-to-tube sheet welds and within acceptable stresses in the tube-to-tube sheet connections.

The combination of a tight fit between the tube and tube sheet, combined with a low stress in the tube-to-tube sheet weld might produce an acceptable product. This also could be an option for the superheater units employing smaller diameter tubes as originally planned for EBR-II. The design of the tube-to-tube sheet connection warrants considerable attention and ingenuity.

A thorough evaluation should be made to establish the size limitation of EBR-II modified units. An extraordinary level of reliability of these units is required to achieve the long-term reliability of liquid metal cooled fast breeder reactor power stations. It must be far superior to that achieved by pressurized water reactors. It warrants greater attention and allocation of appropriate capital cost of a liquid metal cooled fast breeder reactor power plant. These units must achieve extraordinary reliability for economic reasons, not nuclear safety, and should be evaluated on a total economic basis, including the cost of unavailability. The EBR-II design and experience should be included.

PLANT OPERABILITY AND RELIABILITY

EBR-II operated reliably under a broad variety of conditions and missions. Although most of those missions involved operation at full power, many required abnormal operation as compared to conventional power plants. Some involved experiments requiring frequent reactor shutdowns to retrieve information or to examine irradiation experiments. The unique EBR-II refueling system made it possible to shutdown the reactor, make a change in the reactor loading and return to power in less than eight hours. This capability made it possible to operate as a power plant while performing a variety of experiments.

EBR-II operations benefited from the virtually ever-present load that could accept the electric power generated. Also, the 100 percent condenser capability meant that the total power generated by the reactor could be accepted by the system even though it was wasted. The reactor could run virtually whenever it was ready to do so, which was one of the basic objectives of the program. In spite of these somewhat special conditions, EBR-II operated extremely well as a base load nuclear powered generating station connected to a utility power grid.

In the process of evaluating the operational capability of the EBR-II reactor and power system, it was subjected to severe abnormal conditions. Of particular importance are those tests which demonstrated the unique capability of the EBR-II system to survive two intentional loss of cooling malfunctions. The first test involved the loss of coolant flow without scram from 100 percent power level. The second test involved the loss of heat sink without scram from 100 percent power level. In both cases the safety systems were immobilized to prevent reactor scram, and to permit the operation to proceed without intervention. Reactor shutdown and heat removal were accomplished by natural processes.

In the first test, with the reactor operating at full power, the sodium circulating coolant pumps were turned off. Since the reactor continued to operate at power, the reactor coolant temperature increased as the flow rate through the reactor decreased. The reactor fuel temperature increased causing thermal expansion of the fuel and a decrease in the reactivity of the reactor. The reactor has a negative reactivity temperature coefficient. The power level continued to decrease as the temperature increased and the reactor shutdown without operator or automatic intervention. This was a dramatic test and demonstrated a unique capability of this reactor. Equally important, the analysis of this experiment accurately predicted the actual results achieved and provided assurance that the reactor could safely accommodate this severe malfunction. The core temperatures were as predicted and no fuel damage occurred. These results provide confidence in the codes developed to analyze this operation and their applicability to other liquid metal cooled fast breeder reactor systems employing the EBR-II concept.

The second test involved the loss of the heat sink with the reactor continuing to operate at full power (i.e., the secondary system sodium pump was turned off and heat removal from the primary sodium in the intermediate heat exchanger stopped). The uncooled primary sodium exited from the intermediate heat exchanger directly to the bulk volume of sodium in the primary tank raising its temperature. Since the primary system coolant is drawn from this bulk volume, the inlet coolant temperature entering the reactor increased, which caused the reactor temperature to increase. Again the power decreased because of the negative reactivity temperature coefficient of the reactor. This was a somewhat less dramatic experiment because of the extended time required to raise the temperature of the large volume of sodium in the primary tank. Nevertheless, it was an extremely convincing demonstration of the accuracy of the codes used to predict the results. In this experiment, the reactor coolant temperature increased slowly and the power level decreased slowly. In about a half hour, the reactor was essentially shutdown and the bulk primary coolant temperature had increased about 70°F.

These two experiments were conducted on April 3, 1986, and were witnessed by an international and national audience involved with nuclear power safety and reliability. The publishers of *Nuclear Engineering and Design* devoted an entire issue to a series of papers related to these tests and their analysis. (Vol. 101, No. 1, 1987).

During its operating lifetime, a variety of experiments were conducted on EBR-II. Those experiments will be described in another volume. In addition, an extensive history of normal operation of this power plant will be reported including performance and supporting data relevant to nuclear power plant operation.

EBR-II FUEL CYCLE

The ultimate goal of the EBR-II fuel cycle was to demonstrate the feasibility of closed cycle operation of a fast reactor power system operating on the plutonium-uranium fuel system. It was broadened to incorporate on site fuel recycle including the fabrication of recycled fuel components manufactured from highly radioactive, incompletely reprocessed fuel. This experience can be extended to future large power systems in whole, or in part. It may be useful to explore the various options which the operation of the EBR-II fuel cycle present. It would appear that as future systems and processes evolve, at least some of the EBR-II experience will be applicable.

As described earlier in this appendix, some of the EBR-II fuel cycle operations can be directly incorporated into an EBR-III concept. These include the operations involved in disassembling a spent fuel subassembly and separating the fuel elements and other components for appropriate packaging and transfer. Similarly, the capability of assembling a “radioactive new subassembly” (containing plutonium and the complete spectrum of other transuranic elements and perhaps fission products) will be required. Similar capability was demonstrated in EBR-II. Additional technical progress is still needed to process and fabricate recycled fuel elements. EBR-III should be capable of accepting such fuel elements and incorporating them into subassemblies for return to the reactor. EBR-II demonstrated the basic capability to perform these assembly functions well enough that they can be incorporated into the EBR-III concept.

Additional development and demonstration is needed to establish the fuel reprocessing technology and probably some extension of the fuel fabrication and fuel element assembly that was demonstrated in EBR-II.

It is clear that the production capability of the EBR-II Fuel Cycle Facility exceeded the requirements imposed by the reactor. This was primarily the result of the higher than anticipated fuel burnup that was achieved. This capability suggests that one Fuel Cycle Facility may be capable of processing the fuel for more than one reactor and thus enhancing the nuclear park concept.

A second consideration relates to the size of the EBR-II fuel elements and subassemblies. The EBR-II fuel elements were designed conservatively, employing a small diameter cast fuel pin. Future power reactors will almost certainly employ larger diameter fuel elements incorporating larger diameter castings. This should actually simplify the casting operation and increase the size of each casting batch. The other fabrication operations involved, such as assembling the fuel elements and the subassemblies should be easier. It is important to note that essentially the same operations performed to fabricate EBR-II fuel elements would be capable of fabricating larger units for a large power reactor. Also, similar radiation levels could exist and require similar remote controlled operations.

Production requirements and rates would differ, but the basic batch concept employed in the EBR-II fuel cycle lends itself nicely to production flexibility. Although most of this experience is applicable to off-site processing, it is clear that fuel transport, including cooling and shielding, could add complexity to an off-site fuel cycle.

OTHER CONCEPTS CONSIDERED BUT NOT INCLUDED IN EBR-II

The process of evolving the EBR-II concept used the conceptual design of a 150 megawatt electric plant as guidance. In the 1950s, this was considered a “full-size electric generating power plant.” Various ideas were advanced and developed for both the large hypothetical plant and for EBR-II. Some were better suited for the large plant concept and others for EBR-II. Even some of those that were more logical for a large plant were incorporated into EBR-II where feasible and useful. The incorporation into the EBR-II concept of the in situ coolant orificing, already described, is such a feature. It contributed very little to EBR-II performance, but could be a significant asset in larger reactors.

A concept that was not incorporated into the EBR-II design, but appeared to be viable for larger reactors, involved the incorporation of a central blanket. This involved a region in the center of the reactor which would be loaded with depleted uranium, similar to the depleted uranium inner blanket surrounding the core in the EBR-II concept. The arrangement could provide several benefits in a large reactor but was totally impracticable in EBR-II because of its small size. In a large reactor, it could flatten the neutron flux across the reactor. It could minimize the reactivity decrement over time as the plutonium in the core burned out; plutonium would be bred in the center of the reactor with a larger proportionate reactivity effect. This could approach an internal breeding ratio of one, which could be considered an ideal configuration. Finally, this arrangement could enhance the overall breeding ratio of the machine. Although it is less important now, because of the availability of plutonium from weapons programs in the United States and elsewhere, in the long range, breeding will become essential to the full utilization of uranium-238. Therefore, it is described here to encourage future liquid metal cooled fast breeder reactor designers to consider this option for applicability as appropriate at the time.

The central blanket concept as developed at the time consisted of blanket subassemblies comparable to the inner blanket subassemblies in EBR-II. These subassemblies would be positioned in the lower grid/plenum in the same manner and would be supplied with sodium coolant from the high pressure plenum. The grid would be stepped appropriately to control coolant flow to each subassembly to accomplish the desired rate of heat removal. The lower adapter of the subassembly would be provided with appropriate coolant holes to provide the proper coolant flow for its position in the reactor. Because it was determined to be an inappropriate feature in EBR-II, the design and analysis were not completed. For example, it was not determined if a single subassembly configuration could be used for both central and inner blanket subassemblies.

The central blanket subassemblies would require higher coolant flow which might be accomplished by appropriate hole size and spacing in the lower subassembly adapter, in conjunction with appropriate “steps” in the lower plate of the grid/plenum structure. It certainly would be interesting, if it is possible, to design a large liquid metal cooled fast breeder reactor using only two types of subassemblies to configure an entire reactor, plus an appropriate reflector subassembly; and if each type of subassembly would receive the appropriate sodium coolant flow established automatically by its position in the reactor. Further, if this can be achieved, the system might also accommodate the changed requirements arising when subassemblies are “shuffled” to enhance reactor operations. The designers of large liquid metal cooled fast breeder reactor power plants will have an interesting and challenging opportunity, and hopefully, fun in the process; EBR-II was all of these.

Although a central blanket may not be of immediate interest, because there is no need for breeding in the near term, but a “central burner” may be of great interest. This could be an excellent concept for actinide burning and EBR-III could be an “Experimental Burner Reactor-III.” It could be a machine that consumes plutonium to generate power and burn actinides. It should be far more efficient and economical than accelerators for transmuting the long-lived transuranics.

The central blanket feature is included in the description of the evolution of the EBR-II concept, even though it was not used, because it provides not only a long-term option for liquid metal cooled fast breeder reactors, but also a short-term option which could increase their immediate usefulness until their long-term capabilities are needed.

In summary, the EBR-II reactor concept was distorted by the desire to demonstrate the potential to achieve high breeding ratio. It was thought to be theoretically possible to achieve a breeding ratio (actually conversion ratio) of 1.2 with uranium-235 fuel and a breeding ratio of 1.7 with plutonium fuel. At that time, these achievements were thought to be extremely important and worthy of actual demonstration. It would now appear that breeding ratios near unity may be more desirable for at least the intermediate period ahead (perhaps the next century) which could simplify the reactor design significantly. If a less efficient breeding cycle is acceptable, the size of the reactor can be reduced significantly from EBR-II, proportional to power level. In EBR-II, the radial blanket occupies more than 90 percent of the total reactor volume. A much thinner blanket and reflector could be used in larger reactors if greater neutron leakage is acceptable. The EBR-II design and program did not incorporate provisions for exploring the benefits of balancing neutron efficiency verses reactor size, although some experiments involving reflector materials were performed. Also, as described earlier, the possibility of incorporating removable reflector/shielding material at the periphery of the reactor can be considered relative to irradiation damage of the reactor vessel. In EBR-II, two layers of shielding are provided inside the reactor vessel to protect the vessel from radiation damage, but this material is not removable. This limitation resulted from the desire to achieve a high breeding ratio which required a thick blanket. Future designers will have a great deal of latitude to optimize the design of the reactor and its performance.

RETROSPECTIVE

EBR-II represents a radical departure from conventional power reactor design. The unconventional basic concept of a reactor and primary cooling system contained in a large vessel and submerged in a unique coolant such as molten sodium was rather quickly accepted in France, Russia, and Great Britain. Some of the details of this basic concept were altered from the EBR-II design which resulted in quite different plant arrangements. Since a “large EBR-II” has not yet been produced, direct comparisons and evaluations cannot be made. Also, these other concepts have not achieved the same level of experience and success as that achieved by EBR-II.

A major difference involves the directed coolant flow employed in EBR-II, which requires a “cover” over the reactor and piping from the reactor to the heat exchangers to produce a closed coolant system for the high temperature reactor outlet sodium. The merits of this arrangement have been discussed, and they are significant. In retrospect, EBR-II could have demonstrated a much simpler piping arrangement between the reactor and heat exchanger which, perhaps, would have made the hot leg concept more attractive. As described earlier, sodium leakage at the pump outlet was found acceptable; the acceptability of leakage from the reactor outlet sodium piping could have been demonstrated also. As noted earlier, the coolant pressure is much lower at the piping connections.

With respect to the reactor cover, it would appear that increasing the size (diameter) of the cover need not be limiting. It might be prudent to employ three lifting columns rather than two, and the clamping arrangement might require modification or may be avoided entirely, but the EBR-II arrangement should be extendable.

The various liquid metal cooled fast breeder reactor designs that have evolved over the years have employed a variety of fuel handling and transfer concepts. The EBR-II concept was developed to accommodate the unique requirements of the EBR-II fuel cycle, but should be useful even if a different fuel cycle is employed. The EBR-II concept provides great flexibility and excellent reactor availability. As described previously, the operations can be enhanced significantly by the application of technology which has become available since EBR-II was developed. Also, as described, this system can be augmented by a supplementary system for reflector subassemblies and thus enhance the concept.

The cool primary system sodium environment enhances the fuel handling and storage concept. It also enhances the shutdown cooling concept. The expanded heat removal system described previously also benefits from the cool primary system sodium reservoir.

A logical follow on to the EBR-II experience would be to design a larger reactor system, EBR-III, based on the EBR-II design. Then evaluate the details and improve upon them, incorporating all of the applicable experience which has evolved over the intervening years. Retain the EBR-II features unless they prove to be non-expandable.

A final word on steam generators. The EBR-II units approached perfection with respect to reliability. They may be impractical for a large power station. That has not yet been established. If it is necessary to “retreat” from the reliability level incorporated into the EBR-II units, double tubes probably should be the last to go. This feature provides an increased level of reliability to the most vulnerable component in the system, thousands of feet of tubing, which cannot tolerate even the smallest leakage defect. The double tube sheet, of course, provides an additional level of reliability but also an additional level of complexity. A compromise should be evaluated, which incorporates double tubes, but a single tube sheet. Such an evaluation should include the consequences of a potential water/steam to sodium leak at the tube-to-tube sheet weld. A leak at the tube-to-tube sheet weld provides a much more restricted leakage path for the high pressure water/steam, between the tube outside diameter and the thick tube sheet.

The general theme of this book has been that EBR-II is an excellent small experimental liquid metal cooled fast breeder reactor power station which operated extremely well. It demonstrated that such a plant can operate reliably and can do so on some recycled fuel. It demonstrated that the concept of on-site fuel recycle is feasible.

It would appear that the primary deterrent to proceeding with the development of fast power reactors in a normal developmental progression (i.e., DC-3 to DC-4 to DC-6 etc.) is the assumption (or conclusion) that recycling fuels through power reactors is too difficult or uneconomic, or unacceptable for other reasons. This conclusion was probably influenced significantly by the limited merits of “plutonium recycle” in thermal reactors where the benefits are quite limited relative to the increased complications.

Similar requirements may apply to fuel recycle in fast power reactors, but the benefits are tremendously greater. In fact, they are so great that the required technology should be pursued much more vigorously.

As described earlier, the potential capability of this energy conversion system is probably exceeded only by the fusion concept, but with a much higher probability of technological and economic success. Therefore, reality warrants a factual determination (as contrasted to speculative opinion).

This should be accomplished by advancing this concept in a logical, orderly fashion, recognizing that additional technological development and demonstration are needed to place this technology in the “ready” stage. This can be done for fast power reactors sooner and more economically than for fusion reactors and with a much higher probability of success.

This can be done at an acceptable cost and in a time frame that will permit the results to be of value to the development of United States energy policy for the long-term. An EBR-III-type developmental, prototype plant could be used to develop the expanded technology needed and to demonstrate the feasibility, operability, and selectability of this concept for commercial application. Although this facility would not be economically competitive, it could be an extremely valuable investment if the economics are evaluated realistically, the subsidy required could be quite acceptable.

If EBR-III generates about 250 net megawatts of electricity and the reactor core contains 150 fuel subassemblies with an operating life of three years, some rather simple facts can be developed. At 80 percent capacity factor and electricity revenue of 5 cents per kilowatt-hour, the plant would produce gross revenue of about $87 million per year. Since about 50 fuel subassemblies would be recycled per year, the revenue available to produce a subassembly would be about $700,000 per unit at 2 cents per kilowatt-hour of fuel cost, or $350,000 per unit at 1 cent per kilowatt-hour. These funds would be used to recycle the fuel and provide the hardware for one subassembly. These would not be the true costs of the recycled fuel. On a continuing basis, the plant would be transmuting about 500 pounds of plutonium per year. This function has a dollar value that can be established by evaluating its cost when performed by an accelerator (which is being considered). The appropriate transmutation value should be added to the permissible cost of a recycled subassembly, similarly, on perhaps a much longer range basis, the value of reducing spent fuel storage time from thousands of years to hundreds of years should be included.

It would appear that it is premature at this time to reach any conclusion regarding viability (or even technical feasibility) of fuel recycle on a continuing basis (to equilibrium) in fast power reactors. It is also premature, and short sighted to not pursue this matter to a logical conclusion. We will only be able to determine if we should pursue this fantastic unique energy capability after we determine how to do it. There is work to be done before we truly know how to accomplish this. It should have been done some time ago, but still should be done.

There is extensive evidence that energy availability in the 21st century will be critical. The path initiated and pursued through the EBR-II concept has the potential of providing a response to meeting those needs.

The product of this enterprise will almost certainly be needed in the 21st century. It should be undertaken now, to have the results available when needed, and before the experience generated by EBR-II is lost.

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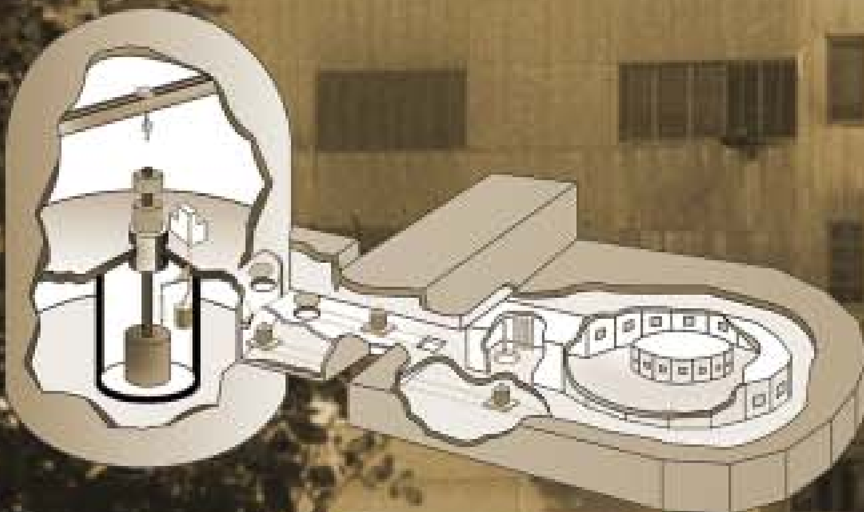
EBR-II Reactor Plant



Fuel Cycle Facility

EBR-II Reactor Core